**Receiving Basin for Offsite Fuels and the Resin Regeneration
Facility Safety Analysis Report, Executive Summary, Chapters 1,
2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17**

by

C. B. Shedrow

Westinghouse Savannah River Company

Savannah River Site

Aiken, South Carolina 29808

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

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ADC & Reviewing Official:	<u>J.N. Dewes</u>
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Date:	<u>9/23/97</u>

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

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EXECUTIVE SUMMARY

This Safety Analysis Report (SAR) documents the safety authorization basis for the Receiving Basin for Offsite Fuels (RBOF) and the Resin Regeneration Facility (RRF) at the Savannah River Site (SRS).

E.1 FACILITY BACKGROUND AND MISSION

Background. The Receiving Basin for Offsite Fuels and the Resin Regeneration Facility at SRS were initially constructed and opened in the 1960s under the management of the site's original Maintenance and Operations contractor, E. I. du Pont de Nemours & Co., Inc. The facility design provided for the safe receipt, handling, storage and shipment of irradiated fuel elements from power and research reactors in the United States. Over the years, the function has not changed, although the mission has expanded to include fuel from SRS and other Department of Energy (DOE) reactors and foreign research reactors.

The facility is located within the H Area of SRS and carries the designations Building 244-H (RBOF) and Building 245-H (RRF).

Mission. The present mission of the RBOF and RRF is to continue in providing a facility for the safe receipt, storage, handling and shipping of spent nuclear fuel assemblies from power and research reactors in the United States, fuel from SRS and other DOE reactors, and foreign research reactors fuel, in support of the nonproliferation policy. Attached to this Executive Summary are overview graphics showing the major foreign and domestic reactor sites and fuel types which may be shipped to the SRS, and the RBOF facility over the next few years. The RBOF and RRF provides the capability to handle, separate, and transfer wastes generated from nuclear fuel element storage. The RRF is equipped to receive resins used in the RBOF and resins shipped from onsite reactor basin facilities, regenerate the resins, and transfer the wastes to the waste tank in the RBOF. The DOE and Westinghouse Savannah River Company, the prime operating contractor, are committed to managing these activities in such a manner that the health and safety of the offsite general public, the site worker, the facility worker, and the environment are protected.

E.2 FACILITY OVERVIEW

The RBOF and RRF are located on a 2-acre site in H-Area, which is within Aiken County, South Carolina. The nearest SRS site boundary to H-Area is approximately 7.5 miles to the west. Paved roads and rail spurs provide transport access.

The facility is located approximately 800 feet to the west of H Canyon and is 139 feet wide by 148 feet long. The RBOF (Building 244-H) encloses six reinforced concrete water-filled basins that extend to a maximum below grade depth of 45 feet. The basins, connected by two transfer

canals, make up a total volume of approximately 500,000 gallons, with the largest basin containing 200,000 gallons. The RRF (Building 245-H) is coupled with Building 244-H and is considered an integral part of the facility. It contains the necessary process and support equipment for regeneration of anion and cation exchange resins from the spent fuel basin facilities.

Fire protection, health protection, industrial hygiene, emergency medical, and security services are all immediately available in H Area.

Chapter 2 of the Safety Analysis Report provides detailed facility description information.

E.3 HAZARD CLASSIFICATION OF RBOF/RRF

The RBOF and RRF is a Hazard Category 2 facility. Hazard Category 2 facilities have the potential for significant on-site radiological or chemical consequences. The facility hazard category was determined in accordance with Department of Energy (DOE)-STD-1027-92, Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports. This categorization provides an initial hazard screening that is used to determine the level of safety analysis that each facility warrants.

In the RBOF and RRF hazard evaluation process, facility segmentation, as defined in each Hazard Assessment Document, was not credited. The hazard evaluation also assumes that certain facility inventories are controlled to limit quantities of radionuclides and chemicals. The results of the Hazard Classification is given in Chapter 3.

E.4 SAFETY ANALYSIS OVERVIEW

The RBOF and RRF activities include safe receipt, storage, handling and shipment of spent nuclear fuel assemblies from power and research reactors in the United States, fuel from SRS and other DOE reactors, and foreign research reactor fuel in support of the nonproliferation policy. The RBOF and RRF provides the capability to handle, separate, and transfer wastes generated from nuclear fuel element storage. This SAR documents the safety authorization basis for managing these activities in a way that poses no unacceptable radiological or chemical dangers to the offsite public, site workers, or the environment.

The following are potential initiating events for environmental releases and worker exposure:

- Fire
- Explosion
- Loss of confinement
- Loss of containment
- Direct radiological exposure

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- Nuclear Criticality
- External hazards (e.g., vehicle or handling accidents)
- Natural phenomena (e.g., seismic, tornado, and wind)

The hazards and accident analyses presented in Chapter 3 of the SAR are the bases for the engineered and administrative controls used to prevent occurrences and mitigate consequences of potential accidents involving radiological and chemical materials. Preventive and mitigating features were identified by reviewing selected facility procedures and historical safety documentation, through meetings with cognizant technical and operational personnel, and by interviewing experienced and knowledgeable facility personnel. Design and administrative features are included as preventors and mitigators.

The main preventors and mitigators included in the safety basis include the passive fuel storage racks and water filled basin systems. In addition to such passive design features, other features that are used to provide layers of defense in depth include installed shielding, use of noncombustible building materials, and concrete curbs around chemical storage areas.

Administrative defense-in-depth is also provided. Examples include training and qualification requirements and use of facility-specific procedures, etc. Chapters 3 and 5 identify and describe in detail all administrative and engineered defense-in-depth measures for the facility, and specifically identify measures enforced by the Technical Safety Requirements.

E.5 ORGANIZATIONS

The RBOF and RRF is managed and operated by the SRS prime contractor, Westinghouse Savannah River Company (WSRC). Within WSRC, the Spent Fuel Storage Division is responsible for managing the facility. The prime subcontractor for construction and construction design support services is Bechtel Savannah River, Inc. The prime subcontractor for security and safeguards support is Wackenhut Services, Inc.

Safety analyses and associated documentation, including this SAR, were completed by the WSRC Projects, Engineering, and Construction Division (PECD). PECD was assisted by Duke Engineering & Services and Science Applications International Corporation in the production of this SAR.

E.6 SAFETY ANALYSIS CONCLUSIONS

The results of postulated accident scenarios are discussed in Chapter 3 of the SAR and presented in Table E.6-1. The consequence analysis for the both the NPH-induced and process-induced criticality events was developed utilizing the 95% confidence methodology. These results have been compared to the evaluation guidelines provided in WSRC Procedure Manual E7. Manual E7 evaluation guidelines are conservative compared to DOE safety goals and radiological

evaluation guidelines, and the highest risk accident scenarios identified in the hazards analysis do not exceed WSRC's evaluation guidelines.

Chemical hazards in the RBOF and RRF were addressed quantitatively in the RBOF and RRF Hazards Assessment Document, Hazards Analysis, and SAR. These analyses indicated that chemical hazards present in the RBOF and RRF may affect facility workers, but do not present a risk to the offsite public. The RBOF and RRF Functional Classification Report, using the results of this SAR, identifies structures, systems, and components, as well as administrative controls, necessary to protect facility workers from the chemical hazards present in the RBOF and RRF.

E.7 SAR ORGANIZATION

This Safety Analysis Report reflects the format and scope guidelines presented in DOE-STD-3009-94. A volume and chapter Table of Contents follows, with a detailed Table of Contents provided in the front matter of each chapter.

E.8 TABLES

Table E.6-1 RBOF Radiological Consequences and Frequencies

Accident	OEP @ 100m (rem)	OEP @ 640m (rem)	MOI (rem)	Frequency (events/yr)
Process-Induced Criticality ^b	4.9E+00	1.60E-01	3.30E-03	6.64E-04
Fuel Rupture	5.30E-01	2.50E-02	3.30E-03	7.23E-01
NPH (earthquake)				
• criticality ^b	3.40E+00	1.10E-01	2.20E-03	2.55E-08
• fuel breach	5.30E+00	2.50E-01	3.30E-02	2.83E-08
• waste tank breach	2.05E-01	6.53E-03	8.10E-04	3.63E-03
Total NPH (earthquake)	2.05E-01 ^a	6.53E-03 ^a	8.10E-04 ^a	
NPH (high winds)				
• criticality ^b	3.40E+00	1.10E-01	2.20E-03	4.01E-05
• fuel breach	5.30E+00	2.50E-01	3.30E-02	3.93E-04
Total NPH (high winds)	8.70E+00	3.60E-01	3.52E-02	
Resin fire	3.40E-02	1.10E-03	1.40E-04	8.15E-04
Resin explosion	3.16E-01	1.01E-02	1.25E-03	2.50E-03
Uncontrolled chemical reaction	5.30E+00	2.50E-01	3.30E-02	2.16E-03

a Radiological consequences do not include events with frequency of less than 1E-06/yr.

b Radiological consequences provided by application of the 95% confidence methodology.

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E.9 FIGURES

SRS Domestic Fuel

Figure E.6-1 SRS Domestic Fuel Receipts

Fuel Receipts

WSRC-SA-11
Rev. 0

MTR
LEU
0.034
13
0.69
0

SITE: Purdue University - IN			
Fuel Type	MTR	MTR	
Enrich	HEU	LEU	
MTHM	0.002	0	
# of Assy.	124	0	
Vol m ³	0.57	0	
Fuel Generation over next 40 yrs. (# of Assy.)	0	22	

SITE: University of Michigan - MI			
Fuel Type	MTR	MTR	
Enrich	HEU	LEU	
MTHM	0.1	0	
# of Assy.	131	0	
Vol m ³	0.75	0	
Fuel Generation over next 40 yrs. (# of Assy.)	0	652	

SITE: Ohio State University - OH	
Fuel Type	MTR
Enrich	LEU
MTHM	0.02
# of Assy.	414
Vol m ³	2.14
Fuel Generation over next 40 yrs. (# of Assy.)	0

SITE: Worcester Polytechnical Inst.-MA	
Fuel Type	MTR
Enrich	LEU
MTHM	0.02
# of Assy.	27
Vol m ³	0.16
Fuel Generation over next 40 yrs. (# of Assy.)	0

SITE: Massachusetts Inst. of Tech. - MA	
Fuel Type	MTR
Enrich	HEU
MTHM	0.031
# of Assy.	68
Vol m ³	0.17
Fuel Generation over next 40 yrs. (# of Assy.)	403

SITE: University of Lowell - MA			
Fuel Type	MTR	MTR	
Enrich	HEU	LEU	
MTHM	0.004	0	
# of Assy.	28	0	
Vol m ³	0.15	0	
Fuel Generation over next 40 yrs. (# of Assy.)	0	28	

SITE: Brookhaven National Labs. - NY	
Fuel Type	MTR
Enrich	HEU
MTHM	0.25
# of Assy.	980
Vol m ³	3.54
Fuel Generation over next 40 yrs. (# of Assy.)	5826

SITE: Rhode Island Nuclear Science Center	
Fuel Type	MTR
Enrich	HEU
MTHM	0.02
# of Assy.	18
Vol m ³	0.1
Fuel Generation over next 40 yrs. (# of Assy.)	130

SITE: Nat. Inst. of Standards & Tech.-MD	
Fuel Type	MTR
Enrich	HEU
MTHM	0.025
# of Assy.	180
Vol m ³	2.16
Fuel Generation over next 40 yrs. (# of Assy.)	690

SITE: Georgia Institute of Technology - GA			
Fuel Type	MTR	MTR	
Enrich	HEU	LEU	
MTHM	0.005	0	
# of Assy.	27	0	
Vol m ³	0.31	0	
Fuel Generation over next 40 yrs. (# of Assy.)	0	133	

SITE: University of Florida - FL			
Fuel Type	MTR	MTR	
Enrich	HEU	LEU	
MTHM	0.01	0	
# of Assy.	258	0	
Vol m ³	0.68	0	
Fuel Generation over next 40 yrs. (# of Assy.)	0	25	

SITE: University of Virginia - VA	
Fuel Type	MTR
Enrich	LEU
MTHM	0.04
# of Assy.	33
Vol m ³	0.19
Fuel Generation over next 40 yrs. (# of Assy.)	12

MTR

Block

HFIR

Spherical Annulus

SRS Foreign Research Reactor Receipts

Figure E.6-2 SRS Foreign Research Reactor Receipts

for Receipts (1996 - 2009)

Notes:

HL "Heavy Load" MTR is an MTR element that contains 500g or greater of U-235 at Beginning Of Life.

Number of shipments for each country are estimates.

HEU - "High Enriched Uranium", >20% ²³⁵U

LEU - "Low Enriched Uranium", 20% ²³⁵U

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 1

SITE CHARACTERISTICS

AUGUST 1997

UNCLASSIFIED DOES NOT CONTAIN UNCLASSIFIED CONTROLLED NUCLEAR INFORMATION	
ADC & Reviewing Official:	<u>J.N. Dewes</u>
J.N. DEWES, Manager, Regulatory Programs	
Date:	<u>9/23/97</u>

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

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ACRONYMS AND ABBREVIATIONS

bpf	blows per foot
CIF	Consolidated Incineration Facility
CPT	Seismic Piezocone Penetration Test
DOE	Department of Energy
DWPF	Defense Waste Processing Facility
EBE	Evaluation Basis Earthquake
EIS	Environmental Impact Statement
ETF	Effluent Treatment Facility
fps	feet per second
FRR	Foreign Research Reactor
RHLWE	Replacement High-Level Waste Evaporator
INEL	Idaho National Engineering Laboratory
ITP	In-Tank Precipitation
ksf	kips per square foot
mph	miles per hour
msl	mean sea level
NEPA	National Environmental Policy Act
NPH	Natural Phenomena Hazard
PC	Performance Category
pcf	pounds per cubic foot
psf	pounds per square foot
RBOF	Receiving Basin for Offsite Fuels
ROD	Record of Decision
RRF	Resin Regeneration Facility
SAR	Safety Analysis Report
SCPTU	Seismic Piezocone Penetrometer Test
SNF	Spent Nuclear Fuel

ACRONYMS AND ABBREVIATIONS (continued)

SPT	Standard Penetration Test
SRFS	Savannah River Forest Station
SRS	Savannah River Site
SSCs	Structures, Systems, and Components
tcf	tons per cubic foot
tsf	tons per square foot
USFS	U.S. Forest Service
UTRC	Upper Three Runs Creek
WSRC	Westinghouse Savannah River Company

1.0 SITE CHARACTERISTICS

1.1 INTRODUCTION

1.1.1 OBJECTIVE

The purpose of this chapter is to provide information that will satisfy the requirements of Department of Energy (DOE) Order 5480.23, paragraph 8.b.(3)(c). This chapter also includes information, if applicable, that will partially satisfy the requirements of DOE Order 5480.23, paragraphs 8.b.(3)(b), (f), and (u) (Ref. 1, 2).

1.1.2 SCOPE

This chapter describes the site characteristics of the Receiving Basin for Offsite Fuels (RBOF) and the Resin Regeneration Facility (RRF) that are necessary to understand the facility environs important to the safety basis. Information is provided to support and clarify assumptions used in the hazard and accident analyses to identify and analyze potential external accident initiators and accident consequences external to the facility. This chapter includes the following (Ref. 2):

- Description of the location of the RBOF and RRF and its proximity to other areas and facilities within the site
- Quantification of those characteristics of the surrounding environment that influence the design, procedures, and safety of the operation
- Identification of the design or evaluation basis of external limits to be examined
- Determination of the degree of suitability of the historical basis for site characteristics in meteorology, hydrology, geology, seismology, volcanology, and other natural phenomena for use in accident analysis
- Specification of population sheltering, population location and density, and other aspects of the RBOF and RRF that affect the surrounding area

When required information is provided in another chapter or in Chapter 1 of the generic Safety Analysis Report (SAR) (Ref. 3), that chapter is referenced to limit repetition. In those cases where policies, programs, and practices important to safe operation are described in detail in other site documents, the salient features are summarized in this chapter and the documents are referenced (Ref. 2).

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1.2 REQUIREMENTS

Westinghouse Savannah River Company (WSRC) documents that specify codes, standards, regulations, and DOE Orders, as well as other site documents that govern program elements addressed in this chapter, are listed in the generic SAR (Ref. 3).

Codes and standards used by the various engineering disciplines in the course of specifying design, operation, maintenance, and inspection criteria are specified in other chapters, primarily the following chapters:

- Chapter 2, "Facility Description"
- Chapter 4, "Safety Structures, Systems, and Components"
- Chapter 5, "Derivation of Technical Safety Requirements"
- Chapter 6, "Prevention of Inadvertent Criticality"
- Chapter 10, "Initial Testing, In-Service Surveillance, and Maintenance"
- Chapter 11, "Operational Safety"

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1.3 SITE DESCRIPTION

1.3.1 GEOGRAPHY

1.3.1.1 Location

The RBOF and RRF are located in the Savannah River Site (SRS), which occupies about 300 square miles (192,323 acres) on the upper Atlantic Coastal Plain of South Carolina. SRS is approximately 25 miles southeast of Augusta, Georgia; 22 miles south of Aiken, South Carolina; and 100 miles from the Atlantic Coast. The generic SAR (Ref. 3) contains additional details describing SRS.

The RBOF and RRF are located on a 2-acre site in H Area and are in Barnwell County, South Carolina (Figure 1.3-1).

The nearest site boundary to H Area is approximately 7.5 miles to the west. Upper Three Runs Creek (UTRC) lies north and Fourmile Branch is south of H Area. Surface drainage from H Area is through tributaries of these two streams. Vegetative cover and soil characteristics and their impacts on erosion and fire hazards are discussed in Section 1.3.1.2 of the generic SAR.

1.3.1.2 Exclusion Area

FACILITY BOUNDARY

Facilities covered by this SAR are listed below. Chapter 2 gives additional details about the facility operation, layout, and arrangement.

- RBOF (Building 244-H)
- RRF (Building 245-H)
- Storage building (Building 244-1H)
- Storage building (Building 244-3H)
- Storage building (Building 244-4H)
- Yard storage

BOUNDARIES FOR APPLYING EVALUATION GUIDELINES

The outer perimeter fence line of the SRS reservation is used as the basis for technical safety requirement limits on the release of gaseous and liquid effluents. Specific evaluation guidelines

were developed for the maximally exposed offsite individual located at the closest point on the SRS boundary (Ref. 4).

There are no specific evaluation guidelines for onsite dose evaluations. Chapter 4 provides discussion regarding the safety of onsite workers for accident scenarios defined in Chapter 3.

Table 1.3-1 provides the approximate distances to the RBOF and RRF from other SRS facilities.

ACCESS CONTROL

The outer perimeter of SRS is fenced; access is controlled by the operating contractor with the assistance of the security contractor, Wackenhut Services, Inc., to restrict access by the public as needed. The roads that pass through or near the perimeter can be blocked by Wackenhut personnel or with the assistance of local law enforcement personnel. Traffic on South Carolina Highway 125 can be restricted rapidly. General access to the plant site, with the exception of public transportation corridors, is limited to badged personnel. Onsite personnel who enter potential radiation areas are provided with dosimeters.

Normal access to the RBOF is through the H-Area gates (fenced area) and is restricted to L- or Q-cleared employees who have a proximity badge. There are no additional security requirements to enter the RBOF and RRF. However, for accountability following facility evacuations, an access and accountability system has been established at the RBOF.

EFFLUENT RELEASE POINTS

Depending on the radioactive content and type of aqueous waste, waste is transferred to either the H-Canyon outside facilities or the H-Area tank farm. Solid waste is sent to SRS radioactive solid waste management facilities for disposal. Gaseous effluents from the RBOF and RRF are discharged through two separate stacks. A 5-foot-diameter, 53-foot-high stack serves the building ventilation system, while a 10-inch-diameter, 53-foot-high stack serves the process vessel ventilation system. Additional details on the RBOF and RRF ventilation systems and waste effluents can be found in Chapters 2 and 9.

RELEVANT SPECIAL FEATURES

No special features of H Area have been noted that would initiate or compound the severity of events analyzed in Chapter 3. Features that would mitigate or reduce the severity of events are described in Section 1.3.1.2 of the generic SAR (Ref. 3).

1.3.2 DEMOGRAPHY

Information on the population within 50 miles of the RBOF and RRF is given in Section 1.3.2 of the generic SAR (Ref. 3). Both permanent and transient populations are discussed. Transient population components investigated are industrial, school, recreational, health care, and casual.

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1.4 ENVIRONMENTAL DESCRIPTION

1.4.1 METEOROLOGY

1.4.1.1 Regional Climatology

Regional climatology is discussed in Section 1.4.1.1 of the generic SAR (Ref. 3).

1.4.1.2 Local Meteorology

Local meteorology is discussed in Section 1.4.1.2 of the generic SAR (Ref. 3).

1.4.1.3 Onsite Meteorological Measurement Program

The Onsite Meteorological Measurement Program for SRS is described in Section 1.4.1.3 of the generic SAR (Ref. 3) and in WSRC-TR-93-0106 (Ref. 5).

Data from the H-Area meteorological tower are representative for characterizing the dispersion climatology of the RBOF and RRF. The H-Area meteorological tower is located at the intersections of Roads E and F approximately 1 mile from the RBOF and RRF (Ref. 5). The Joint Frequency Distribution of hourly averaged wind data from the H-Area meteorological tower for the 5-year period from January 1987 through December 1991 is shown in Section 1.4.1.3 of the generic SAR.

1.4.1.4 Accident Diffusion Estimates

Short-term relative diffusion factors (χ/Q_s) are calculated for use in assessing the consequences of unplanned atmospheric releases. The computer code used for these calculations, AXAIR89Q, is based on the methodology recommended by Nuclear Regulatory Commission Regulatory Guide 1.145 (Ref. 6). A description of this methodology is given in Section 1.4.1.4 of the generic SAR (Ref. 3) and in Section 3.4.1.3.

The χ/Q_s for the RBOF and RRF are based on meteorological data from the H-Area meteorological tower for the period 1987 to 1991. Calculations for unplanned releases conservatively assumed a groundlevel release without any credit taken for the building ventilation stack or the process vessel vent stack.

A limited review of newly constructed facilities was made to evaluate the impact to onsite and offsite receptors from postulated releases from the RBOF. The most limiting structure appears to be the Replacement High-Level Waste Evaporator (RHLWE). The RHLWE is located

approximately 90 meters from the RBOF (Ref. 7). The 100-meter receptor in the downwind direction from the RBOF would, in fact, be positioned within the RHLWE facility. Since the air currents must “negotiate” the presence of the structure by moving above and around the facility, the resultant concentration is lower in the direction of the RHLWE than in other directions at the 100-meter radius. The height and crosswind dimensions of the RHLWE are such that its stack downwash influence is limited to no more than 200 meters from the RBOF; thus, the 640-meter onsite receptor is not affected. Finally, the offsite receptor sees no impact from the presence of structures in H Area, because the wind field conditions have had sufficient travel time to regain “open terrain” behavior.

1.4.2 HYDROLOGY

1.4.2.1 Surface Hydrology

A detailed hydrologic description of the Savannah River and its tributaries is provided in Section 1.4.2.1 of the generic SAR (Ref. 3). Surface waters near the RBOF and RRF include UTRC, Fourmile Branch, Crouch Branch, and McQueen Branch. Surface water is not used for any RBOF or RRF operations.

H Area is located near a water-table divide between UTRC and Fourmile Branch. Near-surface groundwater from the southern part of H Area discharges to an unnamed tributary of Fourmile Branch, approximately 1,000 feet south of H Area. Near-surface groundwater from the northern part of H Area discharges to one of two tributaries of UTRC, which are approximately 1,500 and 4,000 feet north of H Area, respectively. Information pertinent to the Savannah River, UTRC, Fourmile Branch, and other area streams is presented in the generic SAR.

1.4.2.2 Subsurface Hydrology

The water levels, pH, and specific conductance recorded for monitoring wells in H Area are reported quarterly in the Groundwater Monitoring Program report issued by the Environmental Protection Department. Monitoring well data for H Area are reported in the environmental monitoring report for each calendar year. Updated well information, including well locations, is given in the Environmental Protection Department's well inventory report. Information on H-Area water usage; water gradients; permeability, porosity, and flow rates; water chemistry; and potential groundwater recharge areas can be found in Section 1.4.2.1 of the generic SAR (Ref. 3).

1.4.3 GEOLOGY AND SEISMOLOGY

1.4.3.1 Basic Geologic and Seismic Information

Section 1.4.3.1 of the generic SAR (Ref. 3) provides basic geologic and seismic information.

1.4.3.2 RBOF/RRF Geologic and Seismic Information

The specific physiography of H Area is shown in Figure 1.4-1. H Area is located on a divide between Fourmile Branch to the south and UTRC to the north. Ground elevation ranges from about 320 feet above msl along the center of the divide to 200 feet above msl along the southern boundary of the area and 150 feet above msl along the northern boundary near UTRC.

The general surface and subsurface geology underlying H Area may be derived from Section 1.4 of the generic SAR (Ref. 3). H Area primarily overlies strata of the Tertiary-aged Upland Unit with minor portions of the area overlying strata of the Tertiary-aged Tobacco Road Sand.

1.4.3.3 Vibratory Ground Motion

BEHAVIOR DURING PREVIOUS EARTHQUAKES

Behavior during previous earthquakes is discussed in Section 1.4.3 of the generic SAR (Ref. 3).

DESIGN BASIS EARTHQUAKE

The structural design criteria originally used for the RBOF and RRF in 1962 were based primarily on the 1960/61 Standard Building Code and SRS specification 3580 containing Addenda 1 and 2 (Ref. 8) with a seismic design of 0.1g horizontal earthquake force. In 1990, DOE recommended the use of UCRL-15910 (Ref. 9) as guidance for evaluation of the effects of Natural Phenomena Hazards (NPHs) on all DOE facilities. The design basis of Structures, Systems, and Components (SSCs) at the RBOF and RRF for the various Performance Categories (PCs) is now based on DOE Order 5480.28 (Ref. 10) and DOE-STD-1020-94 (Ref. 11).

EVALUATION BASIS EARTHQUAKE - STRUCTURE

Using the guidelines in DOE-STD-1020-94 (Ref. 11), WSRC developed an interim site-specific seismic spectrum for evaluation of ground response for PC-3 facilities like the RBOF and RRF. The background history and description of this Evaluation Basis Earthquake (EBE) are detailed in WSRC-TR-94-0528 (Ref. 12). The interim site spectrum was taken as the envelope of the 84th percentile "distant" (Charleston) earthquake spectrum (described below) and the scaled "local" earthquake (0.19g) of DOE-STD-1024-92 (Ref. 13). This interim EBE was used for the 1995 upgrading of the design of the RBOF and RRF for PC-3 qualification (Ref. 12). This spectrum satisfies DOE-STD-1020-94 and is now incorporated into the SRS Site Standards, WSRC-TM-95-1 (Ref. 14), as presented in Figure 1.4-2, until DOE-STD-1023-95 (Ref. 15) has been implemented for RBOF. The SRS design bases spectra has been developed in accordance with DOE-STD-1023-95 and is documented in WSRC-TR-97-0086 (Ref. 16). This analysis was not available for use in the structural analysis of RBOF but the interim EBE spectra has been

shown to be significantly more conservative (in the 0.5-2.0 Hz frequency range) than the SRS design bases spectra.

EVALUATION BASIS EARTHQUAKE - LIQUEFACTION

The liquefaction analysis described in Section 1.4.3.5 was performed utilizing the distant Charleston event (Ref. 17), which is an 84th percentile rock spectral shape, magnitude (M_w) of 7.5, scaled to 0.13g (Figure 1.4-3).

1.4.3.4 Surface Faulting

Section 1.4.3.4 of the generic SAR (Ref. 3) provides information on surface faulting.

1.4.3.5 Stability of Subsurface Materials

Sediments typically concerned with stability are discussed in Section 1.4.2.1 of the generic SAR (Ref. 3). Detailed analyses have recently been performed to assess the static and dynamic stability of shallow sediments (from ground surface to a depth of 200 feet) beneath a variety of facilities in H Area (Ref. 12, 18, 19, 20). Facility-specific information that can be used to evaluate the subsurface conditions at the RBOF consists of three preconstruction Standard Penetration Test (SPT) borings (244-H-1, 244-H-2, and 244-H-3) and two Seismic Piezocone Penetrometer Test (SCPTU) soundings (244-C1 and 244-C2) pushed as part of the H-Area tank farm investigation (Ref. 20). Depths of the three SPT borings are 101.5, 71.5, and 81.5 feet, respectively. Depths of the SCPTU soundings are 175 and 160 feet, respectively. Figure 1.4-4 provides a plan view of existing subsurface information.

Soft or weak soils generally occur within the Santee Formation and Griffins Landing Member of the overlying Dry Branch Formation. Typically, these zones are associated with carbonate deposits (Ref. 20). The relative depth of these geologic units in H Area is between 110 and 150 feet. These "soft zones" have been encountered in numerous investigations within H Area. Indications of weak soil conditions are derived from SPT N-values less than 10 or SCPTU tip resistances less than 14 tsf. The three preconstruction SPT borings are too shallow to have penetrated into these geologic units. However, the two SCPTU soundings adjacent to the Building 244-H structure that did penetrate the Santee Formation did not indicate the presence of soft zones.

GEOLOGIC FEATURES

Section 1.4.3.5 of the generic SAR (Ref. 3) provides information on geologic features.

PROPERTIES OF UNDERLYING MATERIALS

Properties of surficial sediments in locations near the RBOF and RRF are presented in Table 1.4-1 (Ref. 19, 20). The two SCPTU soundings recently pushed adjacent to Building 244-H and the three preconstruction SPT borings were used to calculate soil parameters including static and dynamic bearing capacity, lateral earth pressure coefficients, and modulus of subgrade reaction (Ref. 21). A typical profile of sediments beneath Building 244-H is presented in Figure 1.4-5. Average soil properties calculated from the two SCPTU soundings at Building 244-H are presented in Table 1.4-2 (Ref. 12). SPT N-values from the three preconstruction borings for Building 244-H are presented in Figure 1.4-6, as compared with In-Tank Precipitation (ITP) SPT N-values (Ref. 12).

The shallow subsurface of H Area has been divided into sublayers based on differences in observed SCPTU measurements and boring data. This engineering stratigraphy divides the shallow subsurface into five basic engineering units (TR1/TR2, TR3/TR4, DB1/DB3, DB4/DB5, and ST), which correspond closely to geologic formations. These engineering layers, as characterized for the ITP and H-Area tank farm (Ref. 20) and compared with Building 244-H, are shown in Figure 1.4-7 and are summarized as follows.

Altamaha Formation (TR1/TR2)

The Altamaha Formation is subdivided into two layers. The upper Altamaha Formation (TR1 layer) is generally more cohesive and less dense than the lower Altamaha Formation (TR2 layer). Soils in the TR1 layer are typically reddish-brown to purple clayey sands (SC) with some mottled purple and brown clay (CH) and silty sand (SM). The soils of the TR2 layer are denser and cleaner than those of the overlying TR1 layer, as seen by the significant increase in tip resistance and decrease in pore pressure generation on the SCPTU logs. The TR2 sediments are dense, silty sands (SM) with interbeds of poorly graded sand (SP), clayey sand (SC), and silt (ML). The TR1/TR2 layer has an average thickness of 45 feet. Tip resistances are moderate (mean $q_c = 93$ tsf) and moderately high friction ratios (mean friction ratio = 4%). SPT N-values in the TR1/TR2 layer range from about 9 to greater than 60 bpf. Shear wave velocities in this layer average 1,320 fps.

Tobacco Road Formation (TR3/TR4)

The Tobacco Road Formation is described based on two layers (TR3 and TR4) but, because of similar material properties determined from laboratory testing (Ref. 20), the TR3 and TR4 layers were later combined into the TR3/TR4 layer. The upper contact with the TR3 layer is defined by a marked decrease in SCPTU tip resistance and an increase in friction ratio and pore pressure. SCPTU signatures of the TR4 layer are similar to those of the TR3 layer but show a slight increase in tip resistance and a decrease in friction ratio and pore pressure generation. This indicates the TR4 layer is denser and more free draining than the TR3 layer. The TR3/TR4 layer averages about 20 feet thick and consists of mottled red, brown, tan, and white clayey sand (SC)

with some interbeds of clayey, silty sand (SC-SM, SM). This layer is recognized on the Seismic Piezocone Penetration Test (CPT) logs as a zone having moderate CPT tip resistances (mean $q_c = 48$ tsf) and moderate friction ratios (mean friction ratio = 3.1%). SPT N-values in this layer range from about 15 to 60 bpf. Shear wave velocities in this layer average 1,118 fps.

Dry Branch Formation (DB1/DB3 and DB4/DB5)

The Dry Branch Formation is divided into layers DB1/DB3 and DB4/DB5. DB1/DB3 consists of the Irwinton Sand Member of the Dry Branch Formation. This layer is typically dense, brown to reddish-brown, poorly graded sand with some silt. The DB2 layer is a 5- to 10-foot thick layer of silty to clayey sand located between the DB1 and DB3 layers that is traceable from the H-Area tank farm. The CPT signature of the DB2 layer indicates a zone of lower CPT tip resistance and pore pressure generation greater than hydrostatic. Because of similar properties, the predominately sandy soils of the DB1 through DB3 layers were combined into a single layer. This combined DB1/DB3 layer averages about 35 feet thick. On the CPT logs, this layer is recognized as a zone of variable, but generally high CPT tip resistances (mean $q_c = 182$ tsf) and low friction ratios (mean friction ratio = 1.1%). SPT N-values in this layer range from about 25 to 60 bpf. Shear wave velocities in this layer average 1,172 fps.

The DB4/DB5 layer comprises the Tan Clay and Griffins Landing Members of the Dry Branch Formation, respectively. In general, the soils in this interval are much more plastic than the overlying Irwinton Sand Member (DB1/DB3) and the underlying Santee/Tinker Formation. Soils in the DB4/DB5 layer range from highly plastic clays, to clayey sands, and to silty sands and sandy silts. Soil logs and core samples show the DB4 layer as a noncalcareous to slightly calcareous, alternately layered brown, tan, and white, clayey sand that interfingers with brown to tan, highly plastic clay. The DB5 layer usually has a lighter color and is more silty than the overlying DB4 layer and is calcareous in some places in the H-Area tank farm. The soils in the DB5 layer are often described as a light olive-green to gray silt to plastic silt with varying amounts of clay and weathered shell fragments. The layer has an average thickness of 20 feet. The CPT logs indicate the DB4/DB5 layer as a zone with variable, generally low CPT tip resistances (mean $q_c = 52$ tsf) and low friction ratios (mean friction ratio = 1.3%). SPT N-values in this layer are not available. Shear wave velocities average 1,061 fps.

Santee and Tinker Formations (ST)

The ST interval includes the Santee and Tinker Formations. In general, it is encountered at a depth of about 100 feet below the natural ground surface. The contact with the overlying Dry Branch Formation is identified on the SCPTU logs as a sharp decrease in the pore pressure. The ST interval is characterized by thin, alternating layers of low and high SCPTU tip resistance. The ST layer averages about 55 feet thick. The ST interval is characterized by thin, alternating layers of low and high CPT tip resistances (mean $q_c = 111$ tsf) and friction ratios (mean friction

ratio = 1.3%). SPT N-values in the ST layer are not available. Shear wave velocities average 1,260 fps.

This unit is highly variable and is rich in carbonate-bearing units including limestones, calcareous sands, and muds. Soft zones are most common in this unit and are identified by low SPT and SCPTU resistances, rod drops during drilling, and drill mud circulation loss. The two SCPTU soundings pushed relatively close to the structure did not indicate soft zones. From mapping of soft zones in H Area, indications are that there are fewer soft zones on the southwestern side of H Area. Thus, it is accurate to state that, based on the available information, analyses for the H-Area tank farm, and engineering judgment, large laterally continuous soft zones do not exist near the RBOF facility.

SOIL AND ROCK CHARACTERISTICS

Properties of surficial sediments in locations near Building 244-H are presented in Tables 1.4-1 and 1.4-2.

EXCAVATION AND BACKFILL

When RBOF was constructed, excavation and backfill performed at the facilities were controlled by duPont Standard Specification SC5E (Ref. 22). All structural fill was to be compacted to a minimum of 95% of the maximum density as defined by ASTM D 1557 (Ref. 23). After 1995, all backfill will be controlled in accordance with individual project specifications.

GROUNDWATER CONDITIONS

A map of the water table at H Area is presented in Section 1.4.2.2 of the generic SAR (Ref. 3). Historically, the water table has been located approximately 30 to 40 feet below grade level at H Area. Updated water-table information is available in the SRS Annual Environmental Report.

RESPONSE OF SOIL AND ROCK TO DYNAMIC LOADING

Soil response to dynamic loading may involve liquefaction or deep subsidence of soft zones. Liquefaction potential is addressed in the following section. Deep subsidence of soft zones is not considered applicable to the Building 244-H structure as evidenced by the existing subsurface information.

LIQUEFACTION SUSCEPTIBILITY

Liquefaction susceptibility analyses were performed using SPT N-values from borings 244-H-1 through 244-H-3 and SCPTU soundings 244-C1 and 244-C2. The analyses were performed for the

controlling earthquake (i.e., the earthquake that produced the highest normalized cyclic stress ratio values in the critical layer). Based on the work performed for the ITP and H-Area tank farm (Ref. 20), this is the 84th percentile motion described in Section 1.4.3.3 (Figure 1.4-3).

The analysis utilizing the SPT N-values was performed in accordance with the methodology given in WSRC-RP-93-606 (Ref. 19), while the analysis using the SCPTU data was performed in accordance with the methodology described in WSRC-TR-95-0057 (Ref. 20). The results suggest the profile at Building 244-H is not susceptible to widespread liquefaction. The total calculated dynamic settlement for all layers above the "Tan Clay" (i.e., above DB4/DB5 layer) is less than about 3 inches (Ref. 17).

EFFECT OF SOFT ZONES AT DEPTH

Soft zones indicated by low SPT N-values from SPT borings, or low SCPTU tip resistances, were not found at the facility, as described in the previous section.

1.5 NATURAL PHENOMENA THREATS

This section identifies specific natural phenomena events considered to be potential accident initiators.

DOE Order 5480.28 (Ref. 10) identifies the responsibilities and requirements to design, construct, and operate DOE facilities so that workers, the general public, and the environment are protected from the impacts of NPHs on DOE facilities. To achieve this policy, the Order has established PCs and target probabilistic performance goals for each category. Performance goals are expressed as the mean annual probability of exceedance of acceptable behavior limits of structures and equipment due to the effects of natural phenomena. Five PCs (PC-0 to PC-4) have been established in DOE Order 5480.28.

An NPH standard, DOE-STD-1020-94 (Ref. 11), was developed from UCRL-15910 (Ref. 9) and provides criteria for the design of new SSCs and for evaluation, modification, or upgrade of existing SSCs so that DOE facilities safely withstand the effects of NPHs. DOE-STD-1020-94 provides consistent criteria for all DOE sites across the United States. These criteria are provided as a means of implementing DOE Order 5480.28 and of complying with DOE safety policy (i.e., SEN-35-91 [Ref. 24] and DOE Order 5480.1B [Ref. 25]) and Executive Order 12699 (Ref. 26) for earthquakes.

Engineering Requirement 01100-02-R (Ref. 27) provides the correlation between the functional classification process and the assignment of Performance Categories (PCs) of SSCs for natural phenomena design criteria and application. Primarily, if an SSC is deemed to be safety class, the assigned PC is PC-3. For safety-significant SSCs, the assigned PC is PC-2 if onsite evaluation guidelines are exceeded, assuming failure of the SSC. At the time of the NPH evaluation for the RBOF, the assigned functional classifications were not known. Therefore, for conservatism, all deterministic evaluations (excluding the chemical pads, tanks, and rain shelter) were performed assuming the SSCs of interest were safety class. This classification requires the SSC to meet the loads associated with PC-3 criteria. The chemical pads, tanks, and rain shelter were assumed to be classified as safety significant thus requiring the SSCs to meet the loads associated with the PC-2 criteria. WSRC-TM-95-1 (Ref. 14) provides the mandatory minimum structural design criteria for all new facilities and modifications to existing facilities, both permanent and temporary, at SRS.

1.5.1 FLOODS

Floods are discussed in Section 1.5.1 of the generic SAR (Ref. 3). The adjacent streams in broad valleys are more than 100 feet lower in elevation than the RBOF and RRF; therefore, flooding from streams will not affect the facility.

1.5.2 EARTHQUAKES

Earthquakes are discussed in Section 1.4.3.

Design basis earthquakes and EBEs are discussed in the generic SAR (Ref. 3). Chapter 2 provides details on the ability of the facility to withstand the PC-3 and PC-2 seismic loads.

1.5.3 TORNADOES

Tornadoes are discussed in Section 1.4.1.1 of the generic SAR (Ref. 3). The annual exceedance probability given in DOE-STD-1020-94 (Ref. 11) for tornadoes is 2×10^{-5} . The fastest-mile wind speed is 137 mph with an atmospheric pressure change of 40 psf at a maximum rate of 20 psf/sec (Ref. 14).

Missile criteria are also specified to account for objects or debris that could be picked up by straight winds, hurricanes, or weak tornadoes. Two tornado missiles are specified as minimum criteria for PC-3. A 2- by 4-foot timber plank weighing 15 pounds is assumed to travel in a horizontal direction at speeds up to 100 mph. The horizontal speed is effective up to a height of 150 feet above ground level. If carried to greater heights by tornado winds, the timber plank can achieve a terminal vertical speed of 70 mph in falling to the ground. A second missile to be considered is a 3-inch-diameter standard steel pipe, which weighs 75 pounds. Design horizontal impact speed is 50 mph; terminal vertical speed is 35 mph. The horizontal speed of the steel pipe is effective up to a height of 75 feet above ground level. Chapter 2 provides details on the ability of the facility to withstand the PC-3 wind loads and missile impacts.

1.5.4 EXTREME WINDS

In accordance with the methodology defined in DOE-STD-1020-94 (Ref. 11), the wind speed for PC-3 at SRS is 109 mph (Ref. 14). The annual exceedance probability given in DOE-STD-1020-94 for straight winds is 1×10^{-3} . A single missile criterion is considered. A 2- by 4-foot timber plank weighing 15 pounds is the specified missile, with a recommended impact speed of 50 mph at a maximum height of 30 feet above ground level. According to DOE-STD-1020-94, straight winds must be evaluated for SRS. Chapter 2 provides details on the ability of the facility to withstand the PC-3 and PC-2 wind loads.

1.6 EXTERNAL MAN-MADE THREATS

This section identifies specific external man-made phenomena associated with the site that are considered to be potential accident initiators, exclusive of sabotage and terrorism.

1.6.1 TRANSPORTATION

Sections 1.6.1.1 and 1.6.1.3 of the generic SAR (Ref. 3) describe the transportation network and the location of airports and airspace within the general area of SRS. A railroad track terminates within the facility, and a roadway provides access for trucks. Potential accident scenarios from surface vehicles within or in the vicinity of the RBOF and RRF are discussed in Chapter 3.

1.6.2 MISSILES AND BLAST EFFECTS

There are no identified scenarios in which energy from nearby man-made sources capable of generating missiles and blast effects could impact the function or operation of safety SSCs in the RBOF and RRF. The risk from explosions is minimized since there are no natural gas lines in use at SRS. The RBOF and RRF are not impacted by loss-of-power events due to the passive safety function of the facility. A diagram of electrical transmission lines is given in Section 1.3.1.2 of the generic SAR (Ref. 3).

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1.7 NEARBY FACILITIES

Land within a 5-mile radius of the RBOF and RRF lies completely within SRS and is used for industrial purposes associated with SRS and as forest land. Forested areas are managed by the Savannah River Forest Station (SRFS), an administrative unit of the U.S. Forest Service (USFS). Through an interagency agreement between DOE and the U.S. Department of Agriculture, the USFS maintains the SRFS to provide timber management, research support, soil and water protection, wildlife management, secondary road management, and fire management.

Other facilities within H Area include H Canyon, H Canyon Outside Facilities, HB Line, Uranium Solidification Facility (USF), Tritium Facilities, Effluent Treatment Facility (ETF), Consolidated Incineration Facility (CIF), ITP, and H-Area tank farm. S Area, the site of the Defense Waste Processing Facility (DWPF), and Z Area, the site of the Saltstone facility, are also located nearby.

The H Canyon contains separation facilities, which dissolve irradiated fuel and target materials and produce solutions containing various products that have been decontaminated from fission products by solvent extraction and ion exchange processes. The H Canyon Outside Facilities, located east of the canyon building, provide feed chemicals and other services to the H Canyon

The mission of the HB-Line facility is to recover actinide products from solid scrap and to produce two different forms of powder. The scrap may come from onsite or offsite sources. It is dissolved in nitric acid and transferred to the H Canyon for reprocessing. One form of powder produced in HB Line is sent to Building 235-F for production of billets. The second form of powder produced in HB Line is sent to Los Alamos National Laboratories, where it is formed into heat source pellets.

The USF is housed within the H-Canyon building. The area occupied by the USF is in Sections 1, 2, and 3 on the first, second, and third levels. USF was designed to receive and process dilute, liquid uranyl nitrate stream transferred from accountability tanks in the A Line. The USF project is currently suspended.

The mission of the Tritium Facilities is to separate tritium from targets irradiated in SRS reactors and returned tritium reservoirs. Tritium Facilities include a new 1-acre underground facility designed to minimize tritium losses to the environment and to reduce waste generation.

The ETF is located on the south side of H Area. The ETF collects and treats routine process wastewater, contaminated canyon facility cooling water, and tank farm stormwater from F and H Areas.

The CIF is located on the east side of H Area. When it begins operation, the CIF will incinerate SRS hazardous, mixed, and low-level radioactive waste.

The H-Area tank farm consists of 29 large underground storage tanks that store high-level aqueous radioactive waste and evaporated saltcake. The ITP and extended sludge processing facilities are located in the tank farm.

The ITP removes radioactivity from dissolved salt solutions by precipitation and then separates the resulting high-activity solids from the decontaminated salt solution via filtration. The high-activity precipitate slurry is stored and transferred to the DWPF. The low-activity decontaminated salt solution will be stored and transferred to the Saltstone facility.

The DWPF is located in S Area, north of H Area. The DWPF converts high-level radioactive waste into an inert solid glass waste form suitable for disposal in an offsite geologic repository.

The Saltstone facility is located in Z Area, north of H Area. The Saltstone facility stores the low-activity decontaminated salt solution from the ITP.

1.8 VALIDITY OF EXISTING ENVIRONMENTAL ANALYSES

As denoted in Chapter 1 of the generic SAR (Ref. 3), a review of existing Environmental Impact Statements (EISs) and Environmental Assessments that address SRS facilities and/or operations indicates that assumptions concerning site characteristics for these documents are the same as those used in the generic SAR. Section 1.8 of the generic SAR discusses the evolution of the site characterization and its use in past National Environmental Policy Act (NEPA) documentation.

As specified in SFS-970013 (Ref. 28), three EISs have been completed with associated Records of Decisions (RODs) issued which are applicable to Spent Nuclear Fuel (SNF) management at the RBOF. In addition, a Notice of Intent has been issued by DOE to assess the course for SNF not previously dispositioned in earlier evaluations. The following discussion summarizes the key aspects of the RODs and Notice of Intent.

The Programmatic SNF Management and Idaho National Engineering Laboratory (INEL) Environmental Restoration and Waste Management Programs Final EIS ROD was issued on May 30, 1995 (Ref. 29). The key SRS impacts from this document are as follows:

- Consolidation of aluminum-based SNF at SRS
- Shipment of aluminum-based fuel elements from U.S. universities, other DOE sites, and other U.S. government sites to SRS, most by calendar year 2014
- Shipment of aluminum-based Foreign Research Reactor (FRR) spent fuel elements to SRS, if receipts of FRR SNF were resumed by the FRR SNF EIS
- Consolidation of non-aluminum-based SNF at INEL
- Shipment of non-aluminum-based SNF inventory at SRS to INEL

The Interim Management of Nuclear Materials EIS RODs pertinent to SNF were issued on December 12, 1995, February 8, 1996, and April 11, 1997 (Ref. 30). The key SRS impacts from these documents are as follows:

- Process Mk31 targets and existing canyon solutions
- Process Mk16/22 SNF plus 81 Taiwanese Research Reactor canisters, one Experimental Breeder Reactor (EBR-II) canister, and miscellaneous SNF and targets
- Process remaining 62 TRR canisters in RBOF

The Nuclear Weapons Nonproliferation Policy Concerning FRR SNF EIS ROD was issued on May 13, 1996 (Ref. 31). The key SRS impacts from this document are as follows:

- Return approximately 17,800 FRR SNF elements to SRS between calendar years 1996 and 2009

- Receive high and low enriched target materials currently in liquid form containing approximately 0.6 MTHM; the original target materials would be similar to the source of sterling forest oxide
- Initiate accelerated development of alternative treatment and/or packaging technologies intended to prepare aluminum-based SNF for repository emplacement; implement this new technology, if possible, by calendar year 2000
- Process any FRR SNF that presents a health and safety concern in F Canyon while canyon dissolution of at-risk materials is in progress, in accordance with the RODs for the Interim Management of Nuclear Materials EIS
- Initiate a study examining the proliferation risks, cost, and timing associated with a decision to chemically separate some of the FRR SNF

The Notice of Intent for the SRS Site-Specific SNF EIS was signed on December 23, 1996, and released to the public in the Federal Register on December 31, 1996 (Ref. 32). The key decisions from this document are as follows:

- Determines the need for and provides NEPA coverage for proposed SNF transfer and storage service, recommended by the Research Reactor Task Team
- Determines disposition of SRS aluminum-based inventories (including new domestic and foreign receipts) which were not designated for processing by the Interim Management of Nuclear Materials EIS RODs
- Supports decision on alternative nonseparative technologies for preparing SRS aluminum-based SNF for geologic disposal

1.9 REFERENCES

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1.10 TABLES

Table 1.3-1 Approximate Distance to the RBOF and RRF from Other SRS Areas

Miles	Area
0.0 - 1.0	200-H Area (other facilities) 200-S Area, Z Area
1.0 - 2.0	200-E Area
2.0 - 3.0	200-F Area N Area
3.0 - 4.0	100-C Area 100-R Area
4.0 - 5.0	700-B Area
5.0 - 6.0	100-K Area 100-L Area 100-P Area
6.0 - 7.0	300-M Area 700-A Area
7.0 - 10.0	400-D Area

Table 1.4-1 Soil Properties at Facilities Near the RBOF and RRF

Property	ITP(Ref. 20)	H-Area Tank Farm(Ref. 20)	Replacement Tritium Facility(Ref. 11)
Saturated Unit Weight	121 pcf	123 pcf	122 pcf
SPT N-Value	27 bpf	28 bpf	16 bpf
Cone Tip Resistance	57 tsf	98 tsf	no data
Friction Ratio	5	3.3	no data
Fines Content	38%	27%	25%
Shear Wave Velocity	1,187 fps	1,151 fps	1,125 fps
Effective Internal Friction Angle	34°	34°	33°
Cohesion	0 tsf	0 tsf	0 tsf

Table 1.4-2 RBOF/RRF Soil Properties Near Foundation Elevation (Ref. 21)

Parameter	Value
Effective Internal Friction Angle	37°
Cohesion	0 tsf
Unit Weight	125 pcf
Active Earth Pressure Coefficient	0.25
At-Rest Earth Pressure Coefficient	0.4
Passive Earth Pressure Coefficient	4.0
Subgrade Reaction Modulus (1- by 1-foot plate)	120 kcf
Allowable Bearing Capacity	
Cask Basin Area	8 ksf
Individual Column Footings along Line E	6.5 ksf
CPT Tip Resistance	93 tsf
CPT Friction Ratio	5%
Shear Wave Velocity	1320 fps

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1.11 FIGURES

Figure 1.3-1 RBOF/RRF Site Location Map and Boundary for Evaluation Guidelines (Ref. 7)

Figure 1.4-1 H Area in the Vicinity of the RBOF and RRF

Figure 1.4-2 Site-Specific Spectra, Horizontal 5% Damping

Figure 1.4-3 Comparison of EBE Rock Spectrum to 50th and 84th Percentile ITP Rock Spectrum

RBOF/RRF Safety Analysis Report

Figure 1.4-5 H-Area Building 244-H Cross-Section

ION A - A'

FOR PHYSICAL DESCRIPTIONS OF THE
ABOVE STRATIGRAPHY REFER TO SECTION
1.4.3.5.2 IN THE SAR.

PLAN VIEW

NO.	DESCRIPTION	DATE	BY	CHKD BY
1	ISSUED FOR CONSTRUCTION	11/11/86	J. L. HARRIS	J. L. HARRIS
2	REVISION	11/11/86	J. L. HARRIS	J. L. HARRIS
3	REVISION	11/11/86	J. L. HARRIS	J. L. HARRIS
4	REVISION	11/11/86	J. L. HARRIS	J. L. HARRIS
5	REVISION	11/11/86	J. L. HARRIS	J. L. HARRIS
6	REVISION	11/11/86	J. L. HARRIS	J. L. HARRIS
7	REVISION	11/11/86	J. L. HARRIS	J. L. HARRIS
8	REVISION	11/11/86	J. L. HARRIS	J. L. HARRIS
9	REVISION	11/11/86	J. L. HARRIS	J. L. HARRIS
10	REVISION	11/11/86	J. L. HARRIS	J. L. HARRIS

Figure 1.4-6 Scatter Plot of the SPT N-value versus Elevation

Elevation	LAYER ITP	Qc avg (tsf)	FR	Vs avg	Qc avg (tsf)	FR	Vs avg	LAYER RBOF
300								
295	SURFACE							SURFACE
290	FILL							
285								
280								
275					93	4.0	1320	UPLAND
270								
265	TR1+TR2	113	2.8	1206				
260	(Upland)							
255								
250								
245								
240					48	3.1	1118	TR3-TR4
235	TR3/TR4	52	2.9	1193				
230								
225								
220								
215					182	1.1	1172	DB1/DB3
210	DB1/DB3	174	0.6	1128				
205								
200								
195								
190								
185	DB4+DB5	49	1.8	1018	52	1.6	1061	DB4/DB5
180								
175								
170								
165								
160								
155								
150	ST	111	1.5	1251	111	1.3	1260	ST
145								
140								
135								
130								
125								
120								
115								

NOTE: Distance between ITP and RBOF is approximately 300 feet.

Figure 1.4-7 Comparison of the Average Properties for ITP and RBOF
(from Cone Penetration Test (CPT) Results)

RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 2

FACILITY DESCRIPTION

AUGUST 1997

UNCLASSIFIED DOES NOT CONTAIN UNCLASSIFIED CONTROLLED NUCLEAR INFORMATION	
ADC & Reviewing Official:	<u>J.N. Dewes</u>
J.N. DEWES, Manager, Regulatory Programs	
Date:	<u>9/23/97</u>

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

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Projects, Engineering & Construction Division
Aiken, SC 29808

SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

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ACRONYMS

ATS	Automatic Transfer Switch
CAM	Continuous Air Monitor
CFR	Code of Federal Regulations
CMU	Concrete Masonry Unit
DOE	Department of Energy
EBS	Expanded Basin Storage
GP	General Purpose
HVAC	Heating, Ventilation, and Air Conditioning
ISO	International Standards Organization
MCC	Motor Control Center
MTR	Material Test Reactor
NCSE	Nuclear Criticality Safety Evaluation
NIM	Nuclear Incident Monitor
NMC	Nuclear Measurements Corporation
NPH	Natural Phenomena Hazard
NSDS	Nuclear Safety Data Sheet
PC	Performance Category
PGA	Peak Ground Acceleration
RBA	Radiological Buffer Area
RBOF	Receiving Basin for Offsite Fuels
RCO	Radiological Control Operations
RRF	Resin Regeneration Facility
RRFC	Research Reactor Fuel Counter
SAR	Safety Analysis Report
SBC	Standard Building Code
SCE&G	South Carolina Electric and Gas Company
SRS	Savannah River Site
SSCs	Structures, Systems, and Components

ACRONYMS (continued)

UBC	Uniform Building Code
UPS	Uninterruptible Power Supply
WAC	Waste Acceptance Criteria
WM	Waste Management
WSRC	Westinghouse Savannah River Company
ZPA	Zero Period Acceleration

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2.0 FACILITY DESCRIPTION

2.1 INTRODUCTION

2.1.1 PURPOSE AND OBJECTIVE

The purpose of this chapter is to provide information that will satisfy the requirements of Department of Energy (DOE) Order 5480.23, paragraph(s) 8.b.(3)(d), as amplified in Attachment 1, paragraph(s) 4.f.(3)(d)4a of the Order, Topic 4 (Ref. 1, 2). The objective of this chapter is to describe the facility and processes to support assumptions used in the hazards and accident analyses.

2.1.2 SUMMARY DESCRIPTION

The Receiving Basin for Offsite Fuels (RBOF) is located in H Area of the Savannah River Site (SRS). The facility is located west of H Canyon and just north of the H-Area tank farm (Figure 2.1-1). The RBOF was constructed in the early 1960s and consists of Buildings 244-H and 245-H and support buildings.

The main structure is Building 244-H, which is a steel framed structure, measuring 139 feet wide by 148 feet long by 45 feet high, that encloses a number of water-filled basins for storage, inspection, disassembly, and repackaging of spent nuclear fuel assemblies.

The building consists of seven main sections separated by partitions and shielding walls. Ventilation systems are provided to exhaust any airborne particulates through filters. The basins, cubicles, and shielding walls are made of reinforced concrete. Most of the above-grade structure consists of standard structural steel shapes with an exterior wall of Transite®. The walls are insulated with Fiberglas® (Ref. 3).

The basin area has an inner wall of Transite® to prevent water damage to the insulation from condensation. The disassembly and inspection basins are separated by an inner concrete wall, and the repackaging basins are enclosed by concrete block walls. The storage basins have stainless steel floors and epoxy-coated walls. Most of the basins have a 0.25-inch-thick stainless steel liner on the floor. The two storage basins are between 22 and 29 feet deep and are filled with a variety of foreign and domestic fuel types.

The purity of the basin water is maintained by pumping the water through a filter deionizer loop. The filter is a Porostone filter, which precedes the deionizer and must be recoated with filter aid when it becomes plugged.

Electrical equipment in Building 244-H is supplied with normal power from the substation and standby power from the normal power system in the H-Area tank farm. Tank farm power is

backed up by a standby generator; therefore, the RBOF will receive either normal or standby power from the tank farm when normal power from the substation is lost.

Building 244-H is equipped with two "once-through" ventilation systems using particulate air filters on the building exhaust and a separate ventilation system for the Radiological Control Operations (RCO) offices and change room. The building is maintained at less than atmospheric pressure so that all air exiting the building is filtered before being released. The basin area is kept at a lower pressure than the rest of the building. After filtration, all air is discharged through a 5-foot-diameter, 53-foot-high stack. The exhaust system for the process vessels in the waste cell and decontamination cell is similar to the building system but is separate and employs acid-resistant components. This system exhausts through a 10-inch-diameter, 53-foot-high pipe.

The Resin Regeneration Facility (RRF), Building 245-H, is a small structure adjoining the west side of Building 244-H. The RRF was built at approximately the same time as the RBOF. Structural details of construction for the RRF are similar to those for the RBOF. The RRF is considered an integral part of the RBOF. The RRF consists of a shielded resin regeneration cell, a control room, and an exterior platform with hose cabinets and trailer-mounted portable deionizers. The purpose of the RRF is to regenerate both cation and anion exchange resin from the RBOF and 100-Area deionizers.

2.1.3 GRADED APPROACH

In accordance with DOE-STD-3009-94 (Ref. 2), the descriptions contained in this chapter reflect a graded approach in the level of detail based on the hazard category, facility complexity, and facility life cycle. Based on the methodology from DOE-STD-1027-92 (Ref. 4), the RBOF and RRF have been classified as Hazard Category 2 facilities (Ref. 5). The level of detail of the description reflects this fact along with the significance of the preventive and mitigative features credited in the hazards and accident analyses. Structures, Systems, and Components (SSCs) considered safety class or safety significant in these analyses are described in greater detail than supporting SSCs.

2.1.4 FACILITY MISSION

The RBOF was constructed in the early 1960s to provide safe storage and handling of a variety of nuclear fuel types. The RRF was built at about the same time and is considered an integral part of the RBOF. The mission of the facility has not changed much since the facility's original construction. The mission of the facility is as follows:

- Receive, handle, and store irradiated nuclear fuel from foreign and domestic reactors, and fuel/targets from onsite reactors
- Repackage nuclear fuel elements and targets into containers or bundles necessary for extended storage and/or shipment to onsite or offsite facilities

- Maintain the provisions to receive resins used in the RBOF and resins shipped from onsite reactor facilities, to regenerate the resins, and to transfer the waste to the tank farm
- Perform each of the above functions safely with respect to the onsite worker, public, and environment through adherence to safety management programs

2.1.5 EQUIPMENT AND SYSTEMS IMPORTANT TO SAFETY

The selection of safety-class and safety-significant SSCs, along with their supporting bases, is documented in Chapters 4 and 5.

2.1.6 CONTENT SUMMARY

This chapter is organized into topic areas, which correspond directly to DOE-STD-3009-94 (Ref. 2). Additional guidance on the content of this chapter is taken from WSRC-TR-94-0466 (Ref. 6). The following are a listing and brief overview of the topics covered in this chapter:

- Introduction, Section 2.1
The purpose of the introduction is to provide an overview of the chapter and to define the objective and scope.
- Requirements, Section 2.2
This section lists the codes and standards that are necessary to establish the safety basis for the RBOF and RRF.
- Facility Overview, Section 2.3
This section presents an overview of the RBOF and RRF, including a brief summary of the processes performed, facility configuration, facility inputs and outputs, mission, and history.
- Facility Structure, Section 2.4
This section describes the layout and structure of the RBOF and RRF. The floor plans are presented and discussed with emphasis on equipment locations and dimensions that are significant to the accident analysis. The materials of construction are also described.
- Process Description, Section 2.5
This section describes the parameters, equipment, and instrumentation and control systems for the RBOF and RRF and the relationship to the SSCs. Information in this section includes the following:

- RBOF radiological and chemical inventories are listed, and the relationship to the processes in the RBOF and RRF are described.
 - Process flows for both the RBOF and RRF are described. Processes described in detail include cask receipt, fuel handling, fuel storage, inspection, disassembly and packaging, shipment, process ventilation, fuel and target cropping, chemical storage pad drainage, cask venting, cask cooling, waste handling, and basin water purification. The resin regeneration process is also described.
 - The instrumentation and control systems for the RBOF and RRF are described. Descriptions include the basin water control system, control room displays and alarms, and the overall control philosophy for the RBOF and RRF.
- Confinement Systems, Section 2.6
This section describes the RBOF/RRF confinement systems. Radiological and chemical confinement systems discussed in Section 2.6 include the basins, facility walls, waste cells, shipping cask, fuel cladding, tanks, pipes, and other process vessels. The RBOF/RRF ventilation system is also described.
 - Safety Support Systems, Section 2.7
This section presents the purpose and an overview of the safety support systems in the RBOF and RRF. Systems described include the Nuclear Incident Monitor (NIM) system, radiation monitors, effluent monitoring, leak detection, waste tank monitoring, and fire protection.
 - Utility Distribution Systems, Section 2.8
This section covers the electrical, water, steam, and air systems used in the RBOF and RRF. Information includes descriptions of offsite sources, principal equipment, and distribution systems. The objective is to present the information in sufficient detail to understand the role that utilities have in the operation of the facility.
 - Auxiliary Systems, Section 2.9
This section describes those features of the facility that are necessary for understanding the operation but do not fit into any of the other sections of this chapter. These features include the breathing air, compressed gas, cold chemicals, cranes and hoists, and communication systems.

2.2 REQUIREMENTS

2.2.1 GENERAL REQUIREMENTS

The RBOF and other 200-Area facilities were originally designed in accordance with the applicable Du Pont engineering standards that were in effect at the time. These standards provided detailed specific instructions and referred to national standards where appropriate. Du Pont standards provided design information and specifications in the following general areas:

- Architectural
- Lubrication
- Civil
- Machine design
- Concrete
- Piping
- Drafting
- Plumbing
- Electricity
- Power
- Environmental protection
- Process equipment
- Fire protection
- Safety
- Heating and ventilation
- Steel
- Instrumentation and control

2.2.2 STRUCTURAL AND MECHANICAL REQUIREMENTS

The original structural and mechanical criteria for the RBOF were based primarily on the 1960/61 Standard Building Code (SBC) and Du Pont specification 3580 containing Addenda 1 and 2 (Ref. 7). Specific requirements included the following:

- Structures shall be capable of withstanding horizontal loads as defined in Section 1205.1 (Minimum Design Loads); 20 psf is the horizontal design load for the RBOF (Ref. 8).
- Exterior walls must be capable of withstanding design loads acting inward or outward. Roofs shall be designed to withstand loads acting outward normal to the surface equal to 1.25 times the horizontal loads specified for the corresponding height zone in which the roof is located.
- Chimneys and stacks shall be capable of withstanding 60% of the moment of stability of the building due to dead loads. The 5-foot-diameter RBOF stack has a design load of 14.4 psf.
- All RBOF reinforced-concrete structures were designed in accordance with ACI 318-56 (Ref. 9).
- RBOF structural steel was designed and constructed in accordance with American Institute of Steel Construction standards.

2.2.3 DESIGN BASIS AND DESIGN BASIS ACCIDENTS

As part of the DOE Order 5480.23 safety analysis upgrades, the RBOF facility was structurally evaluated in conformance with DOE Order 5480.28 (Ref. 10). Specific requirements for demonstrating facility resistance to Natural Phenomena Hazards (NPHs) are included in DOE-STD-1020-94 and DOE-STD-1021-93 (Ref. 11, 12). These requirements are assigned on the basis of the performance categorization for the SSC. The results of these NPH analyses are provided in Section 2.4.1.6.

2.2.3.1 Design/Evaluation Basis Earthquake

The original design was based on earthquake parameters from the 1960/61 SBC and Du Pont specification 3580 (Ref. 7). Specification 3580 required that the RBOF seismic design criteria be evaluated for a horizontal earthquake force equal to 0.1 (total dead load plus equipment weight plus one-half the live load).

In 1982, the seismic criteria for the site was updated (Ref. 13) to define the design basis earthquake as one with 0.20g at Peak Ground Acceleration (PGA). This seismic motion was referenced in earlier RBOF Safety Analysis Reports (SARs) (Ref. 3, 14).

Engineering Requirement 01100-02-R (Ref. 15) provides the correlation between the functional classification process and the assignment of Performance Categories (PCs) of SSCs for natural phenomena design criteria and application. Primarily, if an SSC is deemed to be safety class, the assigned PC is PC-3. For safety-significant SSCs, the assigned PC is PC-2. The assigned PC is based on the result of an NPH event that exceeds the onsite evaluation guidelines. At the time of the NPH evaluation for the RBOF, the assigned functional classifications were not known.

Therefore, for conservatism, all deterministic evaluations (excluding the chemical pads, tanks, and rain shelter) were performed assuming that the SSCs of interest were safety class. This classification requires the SSC to meet the loads associated with PC-3 criteria. WSRC-TM-95-1 (Ref. 16) provides the mandatory minimum structural design criteria for all new facilities and modifications to existing facilities, both permanent and temporary, at SRS.

Depending upon the performance categorization of the SSC, the evaluation basis earthquake for the RBOF is 0.15g (PC-2) or 0.19g (PC-3) Zero Period Acceleration (ZPA).

The deterministic evaluation of the RBOF SSCs of concern is based on the interim site spectrum, which is currently defined as the evaluation basis earthquake as detailed in WSRC-TR-94-0528 (Ref. 17). The interim site spectrum defines a PC-3 seismic event as a 0.19g ZPA event. The RBOF deterministic evaluation provides the basis for which fragility values and associated SSC failure frequencies were derived in Chapter 3. The use of the hazard curve provided in T-CLC-G-00024 (Ref. 18) for determining the occurrence probabilities for incremental seismic events in the RBOF probabilistic analysis is acceptable until new hazard curves are developed for the SRS design basis spectrum (Ref. 19).

Although the SRS design basis spectrum has been developed in accordance with DOE-STD-1023-95 (Ref. 20) and is documented in WSRC-TR-97-0086 (Ref. 21), this analysis was not available for use in the structural analysis of the RBOF nor have the associated hazard curves been developed.

2.2.3.2 Design/Evaluation Basis Tornado and Wind

The original design was based on a wind speed of 63 mph and included a uniform 20-psf load to account for wind effects. Tornado missiles and an atmospheric pressure change were not considered (Ref. 7).

In addition to evaluating the effects of seismic events previously mentioned, the RBOF structure and portions of the RRF were evaluated against the straight wind loads associated with PC-2 or PC-3 wind speeds as required in WSRC-TM-95-1 (Ref. 16). Depending upon the performance categorization of the SSC, the evaluation basis wind speeds are 84 mph (PC-2) or 109 mph (PC-3). The annual exceedance probability for straight winds is 1E-03 and 1E-02 for PC-3 and PC-2 wind speeds, respectively.

For the wind effects from a potential tornado, the RBOF structure was evaluated against the straight wind loads associated with a PC-3 (137 mph) wind speed as required in WSRC-TM-95-1. The annual exceedance probability given in DOE-STD-1020-94 (Ref. 11) for tornadoes is 2E-05. The fastest-mile wind speed is 137 mph with an atmospheric pressure change of 40 psf at a maximum rate of 20 psf/sec.

Missile criteria are also specified to account for objects or debris that could be picked up by straight winds, hurricanes, or weak tornadoes. Two tornado missiles are specified as minimum

criteria for PC-3. A 2- by 4-inch timber plank weighing 15 pounds is assumed to travel in a horizontal direction at speeds up to 100 mph. The horizontal speed is effective up to a height of 150 feet above ground level. If carried to greater heights by tornado winds, the timber plank can achieve a terminal vertical speed of 70 mph in falling to the ground. A second missile to be considered is a 3-inch-diameter standard steel pipe, which weighs 75 pounds. Design horizontal impact speed is 50 mph; terminal vertical speed is 35 mph. The horizontal speed of the steel pipe is effective up to a height of 75 feet above ground level. Tornado wind speeds for PC-2 conditions are not specified in WSRC-TM-95-1.

Section 2.4.1.6 provides the results of the NPH evaluation for the RBOF.

2.2.3.3 Other Design Basis Accidents

Evidence has not been found that severe precipitation, external explosions, fires, pressurization, or other design basis accidents were considered in the original design.

2.3 FACILITY OVERVIEW

This section presents an overview of the RBOF and RRF, including a brief summary of the processes performed, facility configuration, facility inputs and outputs, mission, and history.

2.3.1 HISTORICAL USE OF THE RBOF AND RRF

Construction of the RBOF (Building 244-H) began in 1961. The RBOF is located approximately 800 feet west of H Canyon (Figure 2.1-1). The RBOF was designed to safely receive, handle, and store irradiated nuclear fuel elements. The purpose of the facility, as stated in the original design data (Ref. 15, 22), is as follows:

- Receive, handle, unload, and decontaminate casks of various designs, both horizontal and vertical, containing irradiated fuel elements of types that may be stored in water
- Disassemble fuel elements under water
- Repackage fuel into containers or bundles as necessary for extended storage and/or chemical processing
- Inspect fuel by mechanical and optical means
- Store fuel in racks or containers
- Load fuels into casks of various designs for shipping within SRS or offsite

Over the years, other than upgrading obsolete equipment, no major modifications to the RBOF have occurred. The following upgrades or minor modifications to the facility have been made:

- In 1978, the second cask basin was changed into a storage basin by adding racks and modifications to the 100-ton crane limit switch.
- Heating, Ventilation, and Air Conditioning (HVAC) units and heating coils (except for reheaters) were replaced and a new chiller system was installed in 1994. These modifications made the HVAC systems more efficient by providing better flow rates and by reducing the gaseous and particulate airborne effluents.
- A new waste pump was installed in waste tank EP38 in 1983.
- The sodium nitrate tank was reactivated and converted from using sodium nitrate to using sodium nitrite. This change was made to improve chemistry control (i.e., the sodium nitrite is less caustic than the sodium nitrate).
- Two waste storage tanks, WH-1 and WH-2, were removed.
- The drains for the chemical pads have been rerouted from the 200-H-Area seepage basin to the waste tank.

- In 1996, structural upgrades were made to the Building 244-H superstructure to increase the resistance to PC-3 wind and seismic loads.
- In 1996, modifications were made to Building 244-H to provide upgrades in offices, a women's change room, and a lunch room for RBOF support personnel.
- In 1997, the Expanded Basin Storage (EBS) racks were installed to increase the storage capacity in the #2 storage basin.

The original criteria for the fuel received in the RBOF included fuel enrichment of less than 5% U-235, with one exception in the 20% to 30% range. The facility received a wider variety of fuel types, including natural uranium, enriched uranium, uranium dioxide, uranium carbide, and thorium with cladding of aluminum, stainless steel, or zirconium alloy. Fuels that were metallurgically bonded to the cladding and unbonded fuels were permitted in the facility, including fuel using sodium bonding (Ref. 7).

The RRF (Building 245-H), constructed soon after the RBOF, is located west of the RBOF (Figure 2.1-1). The purpose of this facility was to regenerate cation and anion resins used in the RBOF and onsite reactors. The RRF also cleaned lithium-aluminum target elements before sending them to the Tritium Facilities. The cleaning of lithium-aluminum targets in the RRF ended in the late 1980s. Target cleaning activities were moved to the Tritium Facilities and are no longer performed in the RRF.

2.3.2 CURRENT USES OF THE RBOF AND RRF

The current mission of the RBOF has not changed significantly from the original mission. There have been some changes in the types of materials received in the facility. The fuels currently received and stored at the RBOF are more highly enriched than those specified in the original design criteria. Fuels that have been enriched to as much as 93.5% U-235 are handled in the RBOF (Ref. 7). Due to the associated hazards, sodium bonded fuels are no longer received in the RBOF (Ref. 23).

Over the years, shipping cask designs have been refined and some different types of casks than those originally planned for are now received in the RBOF. Details on the present fuel types and casks that are handled in the RBOF are provided in Section 2.5.

As with the RBOF, the current mission of the RRF is similar to the original mission. Other than the elimination of the target cleaning activities, the processes performed in the RRF have not changed, primarily the regeneration of cation and anion resins used in the RBOF and onsite reactors.

2.3.3 PLANNED FUTURE USES OF THE RBOF AND RRF

SRS has been identified as the site to receive aluminum-clad research reactor fuels from both domestic and foreign consignors (Ref. 24). During this fuel receipt campaign, which is expected to last 15 years, the RBOF will continue to perform its current design mission. However, remaining storage space at the RBOF is insufficient to support the entire campaign. Therefore, the former 100-L-Area reactor basins have been modified to provide the balance of the fuel storage space necessary. Due to special cask handling capabilities, the ability to receive, handle, and ship fuel at the RBOF must be maintained.

The RRF will continue to perform its design function for both the RBOF and 100-Area Reactor Basins. Operation of the RRF is necessary to maintain the deionizing resins, which maintain basin chemistry.

2.3.4 RBOF CONFIGURATION AND PROCESS

The facility consists of Building 244-H, a structure measuring 139 feet wide by 148 feet long by 45 feet high, enclosing a number of water-filled basins that extend to a maximum below-grade depth of 45 feet. A plan view of the facility is shown in Figure 2.3-1. The above-grade structure is fabricated from standard, structural steel members with exterior walls of both Transite® and concrete block. The roof of most of Building 244-H is a stone-ballasted built-up roof system on unprotected steel frame and metal decking. The roof over the inspection, disassembly, and repackaging basins is a 4-inch-thick reinforced-concrete deck with a stone-ballasted built-up roof system. Access to the facility is by means of six doors. A car port door on the north end of the building is provided to allow truck and railcar shipments.

The seven main sections of the facility are separated by walls to provide shielding. An overview of the RBOF process is presented below. Figure 2.3-2 illustrates the process flow for the RBOF. Detailed process descriptions are provided in Section 2.5.

2.3.4.1 Cask Receipt and Storage

The following discussion is generic to most offsite fuel cask receipts that contain unbundled research reactor fuels. Some cask/fuel shipment scenarios require specific unloading instructions.

CAR PORT AND UNLOADING AREA

Casks containing spent fuel are delivered to the RBOF by truck or railcar. Casks delivered by truck may be staged on the parking pads located north and south of Building 244-H. Casks are received into the facility through the rollup door located near the northwest corner of Building 244-H and are positioned in the carport within the working limits of the 100-ton overhead crane. Foreign shipments of fuel and some domestic shipments are delivered inside International

Standards Organization (ISO) containers. If present, the ISO container is opened, receipt inspections and surveys are performed, the cask is detached from the transport vehicle, and overpacks (impact limiters) are removed. The 100-ton crane is used to engage the cask lifting point(s) for transport to the cask wash pit.

CASK WASH PIT

The loaded cask is lowered into the cask wash pit and washed to remove any dirt that may have accumulated in transit. If determined necessary by the receipt survey, the cask will be decontaminated. Deionized water and steam are available to facilitate decontamination. The casks, which may contain high temperature fuel, are vented to relieve pressure through the building exhaust. Venting gases are sampled for airborne contamination. The cask is then filled and flushed using deionized water. The flush water is directed to the sump. Samples are taken of the cask discharge to determine water activity levels. If high activity levels are found, then several additional flushes are performed. If activity levels remain high, the fuel is considered failed and a special procedure is developed for unloading. Some cask lid bolts are removed and the remaining bolts are loosened to ease removal above the cask basin. The cask is then engaged by the 100-ton crane and transferred to the cask basin.

CASK BASIN

The loaded cask is lowered into the cask basin until the remaining lid bolts are at a working height. The lid bolts are then removed and the cask is lowered to the -29-foot-elevation shelf in the cask basin. The cask lid is removed using the 3-ton hoist. The cask basket and fuel configuration are verified to be in compliance with the shipping papers. If no discrepancies exist, then the fuel is unloaded according to procedure. The cask unloading procedure provides detailed instruction for the safe and efficient unloading of fuel. The procedure ensures compliance with the Nuclear Safety Data Sheets (NSDSs) and the Nuclear Criticality Safety Evaluation (NCSE) applicable to the shipment scenario. If allowed by the NCSE, cask baskets containing a safe fuel unit may be unloaded from the cask cavity. Normally, however, only individual fuel units are removed from the cask cavity (one at a time). The fuel element is placed in an unloading bucket to facilitate handling for the purpose of (1) identifying the fuel element, (2) movement to the Research Reactor Fuel Counter (RRFC) for U-235 content measurements, and/or (3) finally movement to a transfer bucket.

For certain casks, the safety analysis authorizes unloading of cask baskets containing up to the safe number of fuel assemblies. The cask basket is placed into a cask basket holder located on the floor of the basin. From there, the fuel is unloaded as individual assemblies as described above. The analysis performed for Chapter 3 determined the probability of occurrence for a criticality event in the unlikely case that a cask basket containing greater than a safe number should need to be unloaded. Such an operation would only be authorized by senior management and under specific request from DOE.

The addition of the RRFC has enabled RBOF personnel to confirm the U-235 content of incoming Material Test Reactor (MTR) fuel. The RRFC (Ref. 25) provides an assay chamber and activation sources (1.2 Ci Am-241 and lithium) to provide an indirect measurement of the U-235 contained in a measured fuel element. The RRFC is normally stored and isolated from other fissile material in the cask basin. Use of the RRFC is directed by a sampling program and implemented by procedures.

The transfer bucket, fitted with a specified insert, provides a level of control on the number of fuel elements as they are unloaded from the cask. The insert ensures that all subsequent handling of the fuel is safe (i.e., total amount of fuel being handled at any point in time is less than or equal to a safe fuel unit as defined by the NSDS). Once loaded, the transfer bucket is used to transfer the fuel elements to a working basin for further processing (cropping, bundling, etc.) or staging. Movement of the transfer bucket to the canal is accomplished by means of a traveling bridge.

Other buckets are also used for unloading, identification, isolation, and other miscellaneous fuel handling/storage activities that do not use inserts. These buckets are typically used to handle only one fuel element or for fuel storage in isolation.

REPACKAGING BASIN

The repackaging basin provides fuel repackaging capability by mechanical means. Fuel elements destined for SRS facilities or offsite shipment are loaded into bundles, packaged, and stored until needed. Preparation of bundles occasionally requires underwater cropping of inert ends from fuel assemblies. Fuel may be staged in this basin as long as the requirements of the applicable NSDS are maintained. Details of fuel cropping are provided in Section 2.5.

#1 STORAGE BASIN

The underwater storage basins provide the means to store varied fuel types in relatively close proximity while maintaining nuclear safety. The #1 storage basin contains four basic rack types:

- Main storage racks, which provide storage for fuel assemblies and bundled fuel elements
- Bucket storage racks, which store fuel contained in reactor sludge buckets
- Test tube storage racks
- Bundle storage racks

Fuel is stored vertically in rows defined by a vertical framework of racks constructed of aluminum I-beams (Figure 2.3-3). Gratings, guide plates, and spacers (referred to collectively as "hardware") are installed between the racks to define the individual storage spaces along the rows. Nuclear safety is maintained through compliance with the spacing limits detailed in the

NSDSs available for each fuel type currently stored in the RBOF. The water level in the basins and transfer canals is maintained by a water level control system described in Section 2.5.

#2 STORAGE BASIN

The #2 storage basin was originally a second cask basin that has been converted to a storage basin by the addition of storage racks. The 9.5- by 16-foot hole in the floor of the basin (originally used for placing the cask) has been covered by a 1/8-inch stainless steel plate on top of a grill that supports the racks and prevents the inadvertent dropping of fuel assemblies into the hole. Fuel is stored vertically in rows similar to the racks in the #1 storage basin. In addition, EBS racks have been installed to store fuel bundles vertically in locations enclosed by aluminum and separated by the framework of the racks. Nuclear safety is maintained through compliance with the spacing limits detailed in the NSDSs available for each fuel type stored in the RBOF.

TRANSFER CANALS

Fuel elements are transferred between basins through a system of canals connecting each basin. Fuel movement through the canal system is navigated by means of the traveling bridge and/or overhead monorail system.

2.3.4.2 Cask Loading and Shipping

Fuel bundles are transferred from the working or storage basins via the transfer canals to the cask basin, where cask loading operations occur. The shipment of fuel is essentially a reversal of the operations previously described. Once the fuel bundles are loaded, the cask lid is put in place, and the cask is lifted to the surface where the lid bolts are tightened. The cask is then transferred by crane to the cask wash pit, and the cask is surveyed and decontaminated as necessary. Finally, the cask is transferred by crane to either a railcar or truck in the car port and secured by means of a cask holddown frame.

2.3.4.3 Disassembly and Inspection

DISASSEMBLY BASIN

The disassembly basin provides the capability to disassemble fuel by mechanical means. Underwater equipment for weighing is also available in the basin. Fuel may be stored in this basin as long as the requirements of the applicable NSDS are maintained.

INSPECTION BASIN

The inspection basin provides the necessary space and equipment for the examination and measurement of fuel. Fuel may be stored in this basin as long as the requirements of the applicable NSDS are maintained.

2.3.4.4 Control Room

Controls for operation of all pumps, valves, and other equipment for purifying and disposing of basin water are located in this room. A 1-foot-thick concrete shielding wall is located between this room and the basin area. This wall provides protection to personnel in the control room from radiation exposure during normal operations.

2.3.4.5 Basin Water Purification

A cell with shielded walls and a removable cover contains the filter and deionizer used for basin water purification.

DECONTAMINATION CELL

The quality of the basin water is maintained by circulating contaminated basin water through the filter deionizer loop system. The decontamination cell is composed of a recycle loop filter and a deionizer. The recycle loop uses tubular type Porostone filter elements to remove contaminants larger than 1 micron. The deionizer contains mixed cation and anion exchange resins that effectively remove both radioactive and nonradioactive ions from the basin water. A recycle booster pump provides the head necessary to flow basin water through the system. One or more basins may be valved into the recycle loop at one time, but total flow through the system is limited by filter pressure drop.

2.3.4.6 Liquid Waste Handling

Adjacent to the decontamination cell is a waste cell that has shielded walls and a removable cover and that contains two waste tanks.

WASTE CELL

The waste cell contains two tanks: waste tank EP37 and waste tank EP38. Waste tank EP37 is a 1,000-gallon-capacity floor and sump drain collection and surge tank, which overflows to waste tank EP38. Waste tank EP38 is a nominal 6,400-gallon-capacity tank used to collect and neutralize liquid waste from Buildings 244-H and 245-H. Depending on the radioactive content

and type of waste in the tank, the waste is transferred to either the H-Canyon outside facilities or the tank farm.

When waste tank EP38 is unavailable or undesired for use, low-level waste trailers are utilized with temporary transfer lines to directly empty the cask wash basin into the low-level waste trailers. These trailers are then used to transport the low-level waste to Building 211-F or 241-F, depending upon the types of waste (Ref. 26).

2.3.4.7 Basement, Offices, and Change Rooms

BASEMENT

The basement area under the main floor of the RBOF primarily serves as a gallery for pipes routed to and from the basins, the decontamination cell, and the waste cell. Various valves located directly under the Building 244-H Control Room are equipped with handles that project through the control room floor. Support equipment, including the deionizer air filter, recycle loop booster pump, and cask wash pump pit, are also housed in the basement.

OFFICES AND CHANGE ROOMS

This section of the building contains the RCO office and counting room, the electrical control room, locker and change room facilities, supervisor's office, and corridors. This section also includes the support facility, which provides expanded facilities for the existing building.

2.3.5 RRF CONFIGURATION AND PROCESS

The RRF (Building 245-H) consists of three major units: a concrete shielding resin regeneration cell, a control room, and an exterior platform with two hose cabinets (Ref. 3). Figure 2.3-1 shows the layout of the RRF. The operations in the RRF are summarized below.

2.3.5.1 Resin Regeneration Cell

The resin regeneration cell consists of three tanks that separate and regenerate the anion and cation resins used to deionize the RBOF basin water, resins from the RBOF, and resins in the portable deionizer from onsite reactor facilities. Resin is transferred between the RBOF and RRF by means of permanently installed piping. Resins shipped from onsite reactor facilities are contained in portable deionizers permanently mounted on truck trailers. Resin may also be regenerated in the portable deionizers by means of hose box stations located outside the RRF.

2.3.5.2 Chemical Pads, Hose Box Stations, and Portable Deionizer Pad

Located at the west end of the facility, hose box No. 1 serves to transfer mixed or unmixed resins to the RRF resin regeneration cell. Hose box No. 2 is used for resin disposal operations. Hose boxes are considered as the two outdoor stations where permanently installed pipes extending from the process tanks inside the RRF terminate. The open ends of these pipes are connected to the flexible hoses which, in turn, are coupled to the portable deionizers from onsite reactor facilities. The resin received from onsite reactors is contained in portable deionizers mounted on trailers, which are parked outside the facility. Several areas in the RBOF and RRF are provided with drainage systems that flow to waste tank EP38, including the hose racks, chemical pads No. 1 and No. 2, the unloading platform, and the waste and resin regeneration cell sumps. These systems are described in detail in Section 2.5.

Chemical pad No. 1 is adjacent to the RRF north of hose box No. 1. Chemical pad No. 2 is located west of the RRF adjacent to hose box No. 2. These areas serve as chemical holding points for RRF operations and waste treatment operations.

2.3.5.3 Supporting RBOF/RRF Buildings**STORAGE AND MAINTENANCE BUILDING 244-1H**

Materials used for the operation of the RBOF facility are stored in this building. RBOF electrical and instrumentation personnel are also stationed in this building.

STORAGE BUILDING 244-3H

Clean and used laundry for the operation of the RBOF facility is stored in this building.

STORAGE BUILDING 244-4H

An assortment of operational support equipment (breathing air hoses, plastic suits, etc.) are stored in this building.

YARD STORAGE

An assortment of hardware used to bundle and store fuel in basins is stored outdoors in racks near the southwest corner of Building 244-H.

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2.4 FACILITY STRUCTURE

This section provides an overview of the RBOF/RRF buildings and structures including basic floor plans, equipment layout, dimensions, and materials of construction. Those features that are significant in understanding the accident and hazards analyses are described.

2.4.1 BUILDING 244-H STRUCTURE

2.4.1.1 Controlling Dimensions for Building 244-H

Building 244-H is a one-story facility constructed on a raised slab on grade level. The irregular-shaped facility measures approximately 139 feet by 148 feet and covers approximately 15,291 ft². A partial basement is located under the control room on the west side of the facility. The basement measures approximately 96 feet by 23 feet, or 2,208 ft². The facility has a steel frame with exterior walls of corrugated cement asbestos panels (north, south, and crane area walls), a reinforced-concrete west wall, and a 12-inch-thick concrete block east wall. Interior walls are unrated and constructed of concrete block or reinforced concrete with extensions of cement asbestos panels at the crane and basin areas. The basement is constructed on a concrete slab with reinforced-concrete walls and roof. The roof of most of Building 244-H is a stone-ballasted built-up roof system on unprotected steel frame and metal decking. However, the roof over the inspection, disassembly, and repackaging basins is a 4-inch-thick reinforced-concrete deck with a stone-ballasted built-up roof system. Similarly, the roof over the #1 storage basin, where the main exhaust equipment is located, and over the office at the north end of the control room, is also a 4-inch-thick reinforced-concrete deck with the same stone-ballasted built-up roof system. Figure 2.4-1 is an elevation view of Building 244-H.

2.4.1.2 Construction Details and Materials (Ref. 7, 27)

The below-grade basins have walls and floors of reinforced concrete. The walls vary in thickness from 2 to 3 feet at grade level to 6.5 feet at 29 feet below grade level. Portions of the basins below grade level extend to a maximum below-grade depth of 45 feet. The basin floors are 3 feet thick. All basin floors, except the #1 storage basin, have a 0.25-inch stainless steel lining. The #1 storage basin floor has a 0.125-inch stainless steel lining. The walls of the disassembly, inspection, and repackaging basins are lined with 0.125-inch stainless steel. The walls of the other basins have an epoxy (Phenoline AE-600®) coating. The basins are provided with wall studs and bolt holes for the installation of underwater equipment. Guard rails surround all basins, and the basins are protected by a 4-inch-high curb. Further details associated with the construction of the concrete basins are available in Reference 27.

The building structure was designed in accordance with the SBC issued by the Southern Building Code Congress International, Inc., and was constructed as a Class III type facility. The structural

steel framing was designed to conform to American Society for Testing and Materials Serial Designation A36 (Ref. 27) with exterior walls of concrete block and Transite®. A 4.5-inch-thick layer of Fiberglas® insulation is provided on the interior surface of walls. The basin area has an inner wall of Transite®, which protects the insulation from condensation.

2.4.1.3 Building 244-H Ground Floor Plan and Layout

Figure 2.3-1 shows the RBOF ground floor plan. Each section is described below.

HIGH BAY AREA

The high bay area incorporates the car and truck port, cask wash pit, and cask basin (all served by the 100-ton bridge crane), and the #2 storage basin. The car port is located just north of the basin room and is 1,701 ft² in area; its dimensions are 21 feet wide by 81 feet long. The floor is 3 feet below the main floor of Building 244-H, and the roof elevation is 23 feet for the east and west bays of the area and 42 feet 9 inches in the center two bays. The south side of the area is open to the basin room, except for the wall between the car port and the maintenance office, which is metal stud and cement asbestos panels to the roof. The east and west walls are 12-inch-thick reinforced concrete to 3 feet high with unprotected structural steel and corrugated cement asbestos panels to the roof. The north side is open to the cask yoke storage area in the center two bays. The west wall has a rollup overhead door for access by railcars and trucks. The north side also consists of two walls of 12-inch-thick concrete to 3 feet high with unprotected structural steel and corrugated cement asbestos panels to the roof.

The cask yoke storage area is north of the car port and is 720 ft² in area; its dimensions are 21 feet wide by 36 feet long, and the area is open to the unprotected structural steel roof at an elevation of 42 feet 9 inches. The south side is open to the car port, and the west, north, and east walls are 12-inch-thick reinforced concrete to 3 feet high with unprotected structural steel and corrugated cement asbestos panels up to the roof. The floor of this section is a concrete slab on grade level with no floor covering.

The storage basin room is 4,612 ft² in area; its dimensions are approximately 45 feet wide by 102.5 feet long. The south wall is corrugated cement asbestos panels on unprotected steel framing, and the west wall is 12-inch-thick reinforced concrete up to 10 feet above the finished floor and then is cement asbestos panels to the roof. The north end of this room is open to the car port. The east wall is 8- by 12-inch Concrete Masonry Unit (CMU) with unprotected openings to the inspection and repackaging basin rooms. The room is open to the unprotected steel framed roof structure at an elevation of 15 feet for the #1 storage basin (southernmost) and 42 feet 9 inches for the remainder of the area.

Car Port and Storage Area

This is a T-shaped area on the north end of the facility. It encloses a miscellaneous storage area on the north wall. The railroad track terminates in this area. Cask vehicles enter the RBOF via the car port rollup door on the west side. Trucks with trailers are positioned outside the car port door and are slowly backed into the building. Trains containing casks are required to stop approximately 200 feet from the car port door, where a derailler is removed from the track. The cask car is then slowly pushed into the building. A car stop is provided on the east side of the building to limit the travel of the cask car (Ref. 28). Floor drains are provided to facilitate washing of cars, walls, or other items for decontamination. Transport vehicles routinely remain in the building only for the duration necessary to complete the cask handling activities.

Cask Wash Pit

The cask wash pit is 12 feet long by 27 feet wide by 3 feet deep. It contains a pit, measuring 15 feet deep by 9.5 feet in diameter, on the west side of the basin for placement of the cask. The cask wash pit floor is lined with 0.25-inch stainless steel, and the walls are covered with an epoxy (Phenoline AE-600®) coating.

Cask Basin

The cask basin is 13 feet long by 27 feet wide by 29 feet deep. The basin has a pit, measuring 45 feet deep by 9.5 feet in diameter, in the center of the basin. Depending on the cask, either the 29-foot floor elevation or the pit is used for placement of the cask. The cask basin floor is lined with 0.25-inch stainless steel, and the walls are covered with an epoxy (Phenoline AE-600®) coating.

#2 Storage Basin

The smaller of the two storage basins was originally a second cask basin and has the same dimensions as the cask basin. The basin was converted to a storage basin by the addition of fuel storage racks. The 9.5-foot-diameter pit in the floor has been covered by a grill to prevent fuel from being inadvertently dropped into the pit. The #2 storage basin floor is lined with 0.25-inch stainless steel, and the walls are covered with an epoxy (Phenoline AE-600®) coating.

#2 Storage Basin Racks

The #2 storage basin racks, like the main storage racks described below, are a vertical framework of aluminum I-beams. The racks are spaced so that there are 11 rows with 12-inch center-to-center spacings (Figure 2.4-2). The effective lateral storage space in each row is 9 feet with the vertical height of the racks extending 18 feet 5.25 inches. One row of rack ties is

provided to interlock the system. Hardware is specifically provided for the #2 storage basin racks and provides for up to 60 storage positions and two tiers.

Six modular racks, referred to as the EBS racks, have been installed in the #2 storage basin to provide the capability to store additional fuel elements in the vertical position (Ref. 29). Each rack consists of 30 aluminum tube sections (6 inches by 6 inches by 12 feet high) in which the fuel bundles are stored. The storage racks provide storage of fuel bundles at a nominal center-to-center separation distance of 7-1/16 inches along a row and at a nominal center-to-center separation distance of 9 inches across a row. The primary function of the tube sections is to maintain spacing requirements between the fuel bundles for criticality safety. During installation of the racks, it was determined that two of the rack positions were closer than the designed spacing. Therefore, a dummy bundle was installed in location T4A (Ref. 30). An EBS fuel bundle may contain up to five MTR fuel elements stored in a tube that is approximately 12 feet long with a 5-inch diameter. Other bundles, such as a General Purpose (GP) bundle or square tube bundle, may be stored in the EBS racks as necessary.

LOW BAY PLATFORM CRANE AREA

This area includes the #1 storage basin south of the high bay area.

#1 Storage Basin

The #1 storage basin is 40 feet long by 27 feet wide with part of the basin having a depth of 22 feet and another part having a depth of 29 feet. The floor of the #1 storage basin is lined with 0.125-inch stainless steel, and the walls are covered with an epoxy (Phenoline AE-600®) coating. The storage basin contains the aluminum racks where fuel elements are stored. These aluminum racks include the main storage, bucket storage, and test-tube storage racks. The layout of the storage racks in the basin is shown in Figure 2.4-2.

Main Storage Racks (Ref. 31)

Table 2.4-1 provides a glossary of terms for the RBOF fuel racks and hardware.

Fuel assemblies and bundled fuel elements are normally stored in the main storage racks in the 42 rows located west of the basin canal. These rows are defined by a vertical framework of racks constructed of aluminum I-beams with a height of 11 feet 2 inches. Gratings, guide plates, and spacers (referred to collectively as “hardware”) are installed between the racks to define the individual storage spaces along the rows (Figure 2.3-3).

The racks are spaced so that there are 21 rows with 9-inch center-to-center rack spacing, 18 rows with 12-inch center-to-center rack spacing, and 1 row each with 25.5-, 16-, and 11-inch center-to-center rack spacings. Rack thickness reduces the width of the row opening by 3 inches;

the hardware, when installed, further reduces row width so that the maximum row opening is about 3.75 inches less than the center-to-center rack spacing.

The effective storage space in each row is 16 feet. The vertical height of the racks is 11 feet 2 inches, but in some cases fuel is stored so that it extends above the top of the racks.

The hardware between the racks serves to both support and space the stored fuel. Up to four tiers of this hardware (depending on the length of the stored fuel) can be installed in each row. Each tier consists of the following:

- A grating that supports and aligns the bottom of the fuel
- Two guide plates near the top of the fuel that center the fuel between the racks
- A number of spacers that bridge the guide plates to position the fuel along the row

Two types of spacers are employed:

- The swing-down spacer pivots on one guide plate to engage a slot in the other guide plate.
- The removable spacer fits into slots cut in both guide plates.

Three rows of rack ties bridge the top of the storage racks. These rack ties pivot on one rack and engage the adjacent rack to interlock and unite the whole rack system. Individual rack ties are raised to permit the movement of a fuel unit along the row and then are immediately re-engaged.

A lock installed at the end of each row bridges the gap between adjacent racks and prevents movement of fuel into or out of the row. The row lock can only be removed by a special tool that is under the control of the building supervisor.

Bucket Storage Racks (Ref. 31)

It is sometimes necessary (or convenient) to store fuel in reactor slug buckets. The bucket storage racks located in the southeast corner of the storage basin provide space for 70 of these buckets. Like the main storage racks, the bucket storage racks are in the form of a vertical framework of aluminum I-beams. There are five rows thus defined. The southernmost row of the system of five-bucket rows was constructed later and differs slightly from the other four. A small removable section is provided in the tier gratings in the row. When this section is removed, a long piece of fuel can be stored in a vertical position extending into the tier above. The spacing of the racks that defines each is such that a bucket stored in any row will be more than 12 inches from a bucket stored in the adjacent row. Vertically, the rows are divided into seven tiers by swing-down gratings that pivot on one rack and engage the opposite rack. The vertical spacing of these gratings is such that the top of a type A, B, or C bucket stored on a lower tier is greater than 12 inches below the bottom of the fuel stored in a bucket in the tier directly above. If a type D bucket, or fuel which extends above the top of the other bucket types,

is stored in a lower tier, an empty space is procedurally required to ensure that the 12-inch separation is maintained. Swing-down spacers are employed to divide each row tier into two bucket storage positions. These are double spacers which ensure that two buckets on the same tier will be no closer than 12 inches from each other and that the outermost bucket will be 12 inches or more from the edge of the rack.

Bundle Storage Racks

Bundle storage racks are installed adjacent (north) to the bucket storage racks and provide four storage rows with 6-inch row openings separated by three rows with 4-inch row openings. Hardware provides 11 to 12 storage positions. The hardware design is similar to the main storage racks. The effective lateral storage space is 5 feet 11.5 inches, and the vertical height is 18 feet 9 inches. Two rows of rack ties bridge the top of the storage racks.

Test-Tube Storage

If the integrity of the fuel cladding is suspect, the fuel is placed in an isolation container (referred to as a "test tube") and stored in a special rack near the basin-to-canal door. Storage capacity is limited to a total of 13 test tubes. Other containers are also used for storage of suspected failed fuel and may be stored in other authorized locations.

The test-tube rack is constructed of aluminum I-beams and is similar to the other storage racks. Four positions are provided for the storage of 14-inch-diameter test tubes. The tubes are stored in a vertical position and positioned by lugs at the base and swing-down spacer plates, which provide 12-inch spacing between tubes.

Three rows are provided for the vertical storage of other tubes. Hardware is used in these rows to establish the storage positions. Either a single tier (for 13-foot 2-inch tubes) or two tiers (for 5-foot 1-inch tubes) can be used. Three tubes can be stored in each tier.

Swing-down spacers, tier gratings, and rack spacing provide 12-inch spacing between storage positions. The storage arrangements can be modified, if required, by substituting different hardware such that fuel bundles or elements in other than test tubes can be stored in this rack.

LOW BAY OPERATING BASINS

This area includes the transfer canals and the inspection, disassembly, and repackaging basins. The inspection and disassembly basin room is 1,830 ft² in area; its dimensions are 30 feet wide by 61 feet long, and the room is open to the roof at 15 feet. The south wall is 8-inch CMU, and the west end of the room is 8-inch CMU with a 7-foot-wide unprotected opening into the storage basin area. The north and east walls are 12-inch CMU. All of the walls extend to the roof. The repackaging basin room is located south of the inspection and disassembly basin room. It is 935 ft² in area; its dimensions are 17 feet wide by 55 feet long, and the room is open to the roof.

at 15 feet. All of the walls of the room are 8-inch CMU to the roof. The west wall has a 7-foot-wide unprotected opening to the storage basin area. A single exposed cable tray passes through both rooms along the west wall just below the roof elevation. This cable tray runs north to south and passes through the south wall of the repackaging basin room, the common wall between the two rooms, and the north wall of the disassembly basin room. None of the wall penetrations have fire barrier seals. The floor of these rooms is concrete slab on grade level with no floor covering. The storage basins are connected to the three basins in these two rooms by canals in the floor that are covered by open metal grating.

Transfer Canals

The #1 and #2 transfer canals are 3 feet wide and 29 feet deep and connect the working basins together. With the exception of portions of the canal outside of the disassembly and inspection basins, the transfer canals are covered with removable gratings. The floor and walls of the transfer canals are covered with an epoxy (Phenoline AE-600[®]) coating.

The original design specified seven underwater doors, which served to compartmentalize the basin transfer canal system into eight sections. These doors are no longer used to isolate individual basins; therefore, Chapter 3 did not consider the failure of these doors and the corresponding accident consequences.

Part of the #1 transfer canal is referred to as the harp flush area and is used to store the standard fuel assembly for the RRFC. This area is located in the canal at the opposite end from the entrance into the inspection basin.

Inspection Basin

The inspection basin is 5 feet wide and 30 feet long. Ten feet of the length next to the door is 29 feet deep and is provided with a centrally located pit 3 feet in diameter with the floor at the -44-foot elevation. The remaining length of the basin is 14 feet deep. The inspection basin floor is lined with 0.25-inch stainless steel, and the walls are lined with 0.125-inch stainless steel. Recessed sockets are provided in floors and walls for equipment anchor bolts.

Disassembly Basin

The disassembly basin is 5 feet wide, 40 feet long, and 14 feet deep. A 6-foot-long rectangular pit with the floor at the -19-foot elevation is centrally located in the basin. A hopper is located in the disassembly basin to collect shavings from the tube cutter and is provided with 1-1/2-inch schedule 40 pipe to allow the debris to be transferred to a suitable location (Ref. 32). The disassembly basin floor is lined with 0.25-inch stainless steel, and the walls are lined with 0.125-inch stainless steel. Recessed sockets are provided in floors and walls for equipment anchor bolts.

Repackaging Basin

The repackaging basin is 5 feet wide and 46 feet long. The initial 15 feet from the canal door is at the -29-foot elevation with a centrally located 3-foot-diameter pit extending down to the -44-foot elevation. The next 25 feet has a floor at the -15-foot elevation and the final 6 feet has a floor at the -9-foot elevation. The repackaging basin floor is lined with 0.25-inch stainless steel, and the walls are lined with 0.125-inch stainless steel.

ELECTRICAL EQUIPMENT ROOM/OFFICES

The southeast corner of Building 244-H contains an electrical equipment room, the RCO operations office, access corridors to the operations areas, and the men's toilet and change room. A small part of the RCO operations office is a Radiological Buffer Area (RBA). This section of Building 244-H consists of 12 rooms and corridors that serve support functions for the storage areas and RBAs of the RBOF complex. The walls in this support area consist mainly of the CMU type.

Support Facility

The Support Facility (Ref. 33) is located adjacent to the southeast corner of the main building and is an "L"-shaped pre-engineered steel structure, approximately 82 feet long on the south wall and 90 feet long on the east wall. It was completed in 1996 and provides offices, a women's change room, and a lunch room for RBOF support personnel. The Support Facility is considered a separate structure for fire hazards analysis and originally was classified as a General Use Facility. The exterior wall of the older office section of Building 244-H was upgraded to ensure that the structures were considered as separate fire areas as defined in DOE Orders. The expansion added approximately 4,800 ft² of floor area.

As part of the RBOF wind load evaluations, the RBOF Office Building Extension design package was reviewed to determine if the building satisfied the 100-mph wind speed as defined in the SBC. According to ECS-CES-95-0085 (Ref. 34), the pre-engineered office building adequately supports the performance goals for the modified RBOF building. Failure of this building does not result in failure of the interior 12-inch Column line D and interior 8-inch Column line 4.8 CMU walls.

CONTROL ROOM

The control room is 1,640 ft² in area; its dimensions are 20 feet wide by 82 feet long by 8 feet high to the metal egg-crate ceiling. The roof is an unprotected structural steel assembly at 15 feet above the floor. The floor of the control room is a 24-inch-thick reinforced-concrete slab over the basement. The south wall is cement asbestos panels on unprotected steel framing. The south 15 feet of the west wall is cement asbestos panels on unprotected steel framing, and the

remainder of the wall is 36-inch-thick reinforced concrete. The north wall is metal framing with cement asbestos panels on both sides. The east wall is 12-inch-thick reinforced concrete to 10 feet above the finished floor and metal framing with cement asbestos panels to the roof.

The supervisor's office is located on the west wall of the control room. It is 128 ft² in area; its dimensions are 8 feet wide by 16 feet long by 8 feet high to the cement asbestos board ceiling. The floor is a 24-inch-thick concrete slab over the basement. The west wall is 36-inch-thick reinforced concrete, and the north, east, and south walls are metal stud with cement asbestos panels on both sides.

The maintenance office (formerly the instrument shop) located at the north end of the control room is 266 ft² in area; its dimensions are 14 feet wide by 19 feet long by 8 feet high to the metal egg-crate ceiling. The floor of this office is a 24-inch-thick reinforced-concrete slab over the basement. The south and north walls are metal framing with cement asbestos panels on both sides; the west wall is cement asbestos panels on unprotected steel framing; and the east wall is 12-inch-thick reinforced concrete to 10 feet above the finished floor and metal framing with cement asbestos panels to the roof. The roof is a continuation of the control room roof.

2.4.1.4 Building 244-H Basement Floor Plan and Equipment (Ref. 35)

The basement area of Building 244-H serves primarily as a pipe gallery and service area for the process equipment on the ground floor. Access to the basement is by means of a stairway in the maintenance office and a second stairway at the south end of the control room. The accessible areas of the basement are limited to the pipe gallery and service area beneath the control room (Figure 2.4-3). The cask wash pit pump (EP6) and the recycle loop booster pump (EP13) are located in the basement below the control room.

BASEMENT

The basement service area is located under the control room. The area is divided into three main areas: a vestibule at the north end, the center service area, and the pipe alleys that run around the outside of the service area. The center service area is 556 ft² in area; its dimensions are 9.5 feet wide by 58.5 feet long by 8 feet high. The basement floor slab is 10 feet below the floor level of the control room. The vestibule is 162 ft² in area; its dimensions are 9 feet wide by 18 feet long by 8 feet high. The pipe alley is a corridor that runs around the outside of the basement area and serves as a pipe access space. The basement is constructed of reinforced concrete. The outer walls are 24, 36, and 52 inches thick, and the roof is 24 inches thick. The control room floor is a slab on grade level. The interior walls between the service area and the other areas are 12-inch-thick reinforced concrete. Access to the basement is by a stairway from the maintenance office or by a stairway at the south end of the control room.

DECONTAMINATION/WASTE CELL

This section of Building 244-H consists of the two support cells for the RBOF storage basin water systems. The waste and decontamination cells are separated from the Building 244-H Control Room by a 36-inch-thick concrete wall and also from the Building 245-H resin regeneration cell by a 36-inch-thick concrete wall. The two cells are located west of the control room.

The waste cell is 420 ft² in area; its dimensions are 15 feet wide by 28 feet long by 39 feet high. The floor slab is 24 feet below the normal floor level of Building 244-H. It is constructed with 36-inch-thick reinforced-concrete walls, a concrete slab floor, and an unprotected steel framed roof. The only access to the cell is by removing the north cell cover hatch on the roof. The cell is a restricted access radiation area that is normally unoccupied.

The decontamination cell is located just south of the waste cell, separated by a 36-inch-thick reinforced-concrete wall. The cell is 400 ft² in area; its dimensions are 15 feet wide by 26.6 feet long by 18 feet high. The floor of the decontamination cell is 3 feet below the normal floor elevation of Building 244-H and is a 24-inch-thick reinforced-concrete slab over the basement of Building 245-H. The walls are 36-inch-thick reinforced concrete, with a radiation shielding "L"-shaped entry in the southeast corner of the cell. The shielding arrangement is specified such that, in the event of a release of up to a million curies in any basin, and after the assumed evacuation of the building, the water in that basin can be flushed to an underground waste tank by displacement with clean deionized water without requiring exposure of personnel to radiation exceeding 5 rads/hr or to an accumulated dose exceeding 1 rad for any individual (Ref. 7).

2.4.1.5 Building 244-H Roof Plan and Layout (Ref. 36)

Access to the Building 244-H roof is by means of outdoor stairways located east of the cask unloading area and west of the decontamination cell. These stairways connect with a walkway that provides personnel with access to the roof area. The largest portion of the Building 244-H roof is at the 15-foot elevation. This roof section is above the disassembly, inspection, and repackaging areas; the control room; and the waste cells. The roof above the waste and decontamination cells includes four hatches for removal of contaminated equipment from the cells if necessary.

The heating and ventilation units for the control room and the waste cells are mounted near the north end, over the control room on the 15-foot-elevation roof section. Using stainless steel tubing, these units feed the building's main exhaust system, which is mounted on a concrete pad near the south end. Equipment in this area includes the exhaust fan, filters and prefilters, and the 53-foot exhaust stack.

Equipment mounted near the east section of the 15-foot-elevation roof includes air conditioning units for the basin and the office areas.

The roof section above the unloading, cask, and storage basins is at an elevation of 42 feet 9 inches. The recycle tank (EP48) and the deionized water head tank (EP49) are mounted in the northeast corner of this section. Three 20- by 20-inch exhaust ducts tie into a plenum, which runs south to the filters on the 15-foot-elevation roof.

The east and west sides of the roof above the unloading area are at an elevation of 23 feet. No major equipment is located in this area. A small section of roof at an elevation of 12 feet is over the locker room, electrical control room, and office area. Other than several ducts, no equipment is mounted in this area.

2.4.1.6 Results of NPH Analysis for Building 244-H

For the NPH evaluation relative to Building 244-H, the following SSCs and corresponding NPH events were analyzed (Table 2.4-2):

- Concrete-reinforced storage basins - seismic loads and loads resulting from building failure
- Building superstructure - seismic and wind loads
- CMU walls - seismic and wind loads
- 100-ton crane and traveling bridge - seismic and wind loads
- Fuel storage racks - seismic loads

WIND AND TORNADO EVENTS

During initial NPH evaluations, the RBOF steel superstructure and CMU walls were found to be marginally capable of withstanding the 84-mph PC-2 wind. Specifically, the CMU walls on Column A (between Column lines 6 and 7) were suspect at wind speeds greater than 62 mph. Loss of these walls could cause an increase in the internal pressure of the building, adding loads to the other highly stressed walls and potentially resulting in progressive failure. On this basis, modifications were made to the RBOF facility (Ref. 37) to increase the wind resistance of the steel superstructure to meet the PC-3 wind speeds.

With the above modifications incorporated, the resistance of Building 244-H was significantly improved (see Table 2.4-2). The following scenarios are based on the median failure (50% probability of failure) results determined in the fragility analyses for the structure when it is exposed to extreme winds or tornadoes (Ref. 38). Wind-driven missiles are not included in these failure scenarios. If included, missiles could cause breaches in the exterior cladding systems of the building, resulting in premature two-sided differential pressurization of the internal elements.

- 119 mph: The wind load on the girts and siding results in the postulated failure of the high bay girts (on Column line 1 or 6) in the outwards direction. This failure causes a negative pressure internally for the building. The negative pressure causes a

differential pressure on the windward side of the high bay siding and girts due to the wind pushing inwards in addition to the negative internal pressure. This combined differential pressure causes the windward siding and girts to fail inwards. The critical wall becomes the 8-inch CMU wall on Column line 4.8, which threatens the repackaging and disassembly basins. However, at this point, the CMU wall is exposed to only a one-sided pressure and continues to be protected from a two-sided differential pressure by the other CMU walls.

- 136 mph: The RBOF Office Building Extension is postulated to collapse due to the loss of its wall diaphragms. The loss of the RBOF Office Building Extension results in direct wind pressure on the unmodified 12-inch exterior CMU walls on Column line A and on Column line 7. These 12-inch exterior CMU walls are postulated to be breached at this point. This breaching would result in the loss of support for the roof members of the low roof between Column line 6 and Column line 7. The resulting collapse of the low roof would then lead to damage of the unmodified 12-inch Column line 6 wall. This could cause a full two-sided differential pressure on the interior 12-inch Column line D wall, which threatens the storage basin and the interior 8-inch wall on Column line 4.8, which threatens the repackaging basin. However, at 136 mph, this internal differential pressure is not sufficient to cause the failure of these walls. Therefore, the internal walls remain stable.
- 191 mph: The next point of failure postulated is the failure of the vertical bracing on Column lines G and E. The loss of the bracing causes the steel structure to become unstable and collapse. This failure point is postulated to occur at approximately the same time the block walls discussed above have been exposed to a two-sided differential pressure of sufficient magnitude to cause the walls to fail. The postulated failure at 191 mph is complete for the above-grade structure.

The loads associated with complete failure of the Building 244-H superstructure are not sufficient to cause any major damage to the concrete basins; therefore, the effects of wind-induced basin failures are not analyzed.

SEISMIC EVENTS

Building and Basins

The following scenarios are based on the median failure results determined in the fragility analyses for the structure when it is exposed to seismic events (Ref. 38). There are two potential failure mechanisms for the concrete basin during a seismic event: (1) basin failure due to ground acceleration, and (2) basin failure due to differential or liquefaction settlement of the soil following a seismic event. Soil liquefaction and fuel rack failure effects are not included in determining the following fragility values:

- 0.43g: At this point, the main building bracing is at its median failure capacity and it is postulated that the braces fail. As each brace fails, it throws its load to other parts of the lateral force resisting building frame. These parts are already loaded to near capacity and will not be able to accept additional loads. Thus, the failure is postulated to be progressive, resulting in the total collapse of the above-ground structures.
- 2.11g: This is the median capacity limit for the basin walls. The postulated failure is wall buckling and inwards collapse due to the soil loads on the outside of the basin walls. The resulting volumetric changes on the basins would cause the contained water to be forced out over the top edges of the basins as well as allow the water to backflow out into the surrounding soil through large gaps in the walls. This limit does not consider the effects of soil liquefaction.

As noted in Section 1.4.3.5, the geologic profile at Building 244-H is not susceptible to widespread liquefaction. The total calculated dynamic settlement for all layers above the "Tan Clay" (i.e., above DB4/DB5 Layer) is less than about 3 inches. The effects of dynamic or liquefaction settlement on the concrete basins and the potential for basin draindown are discussed in Chapter 3.

Fuel Racks

The seismic evaluation of the RBOF fuel racks (Ref. 31) included the structural components (I-beams, floor supports, wall supports, fuel storage hardware, and fuel bundling tubes) located in the #1 and #2 storage basins. The fuel racks, which have been qualified to PC-3 criteria, in the #1 storage basin are the main storage racks, bucket storage racks, bundle storage racks, and test-tube storage racks. The racks in rows YA through YK of the #2 storage basin, which also have been qualified to PC-3 criteria, are similar to the main storage racks; however, the hardware is different (Ref. 31). Also, the #2 storage basin contains the EBS vertical tube storage racks, which have been qualified to PC-3 criteria (Ref. 39).

The seismic response of the fuel and the fuel racks was calculated by the finite element analysis technique. This technique is applied in two steps that use a linear 3-D full pool model and a nonlinear single fuel model. The linear 3-D full pool model produces the overall response of the fuel racks, which includes stress distribution, support reaction, and response frequency. The nonlinear single fuel model is compiled from fuel rack effective structural properties, extracted from the linear 3-D full pool model, and a fuel assembly model with a structural gap connection between the fuel and the top of the fuel rack. The nonlinear single fuel model produces the response due to the structural gap of the fuel assembly.

A key input to the linear and nonlinear models is the weight of the fuel stored in the racks. This input is specified in Tables 2.4-3 and 2.4-4 under the column "Analyzed Fuel Weight per Row" and represents the values defined in ECS-SSG-96-0028 and EFR-RSE-96-0115 (Ref. 40, 41). The listed fuel weights are representative of the fuel stored in the RBOF during June 1996.

The governing weight used in the analysis is dependent upon the direction of the loading. Since the fuel racks are connected in the north/south direction by rack ties and storage hardware, the load from all the storage rows is transmitted through the structure to the pool wall. The margin to code for this structure is more closely related to the total fuel weight stored in all the racks than the fuel weight stored in each row. Thus, the margin to code for the structure, which is governed by the north/south loading, is based upon the total weight of fuel stored in all the storage racks.

Since the fuel racks are not significantly connected in the east/west direction, the load from each rack is transmitted to the pool floor supports by each fuel rack frame. Thus, the margin to code for the east/west seismic loading is based upon the weight of fuel stored in each row.

Therefore, the fuel rack capacity for stored fuel weight must address the capacity of fuel stored per row and the capacity of the total weight of fuel stored in all the fuel racks.

T-CLC-H-00271 (Ref. 31) evaluated the fuel rack and fuel storage hardware stresses due to the average weight of the stored fuel and also addressed the storage hardware stresses produced by the actual stored fuel weight in the main storage racks. The actual weight of the fuel stored per row is compared with the average fuel weight per row to determine the locations where the fuel weight exceeds the average weight used in the analysis. For the locations that exceed the average fuel weight, the maximum capacities of the fuel rack and fuel storage hardware are calculated and compared with the actual fuel weight to determine the actual stresses and margins to code.

The structural capacity of the fuel rack I-beams is calculated for all the fuel racks by using the value of the most highly stressed component in the racks. The most highly stressed component for the east/west seismic loading is a horizontal I-beam, and the most highly stressed component for the north/south seismic loading is the rack tie support block weld.

The fuel rack structural capacities and the fuel storage hardware capacities are combined with the values of specific fuel weight per row to produce the structural capacity and margin to code for the racks, as shown in Tables 2.4-3 and 2.4-4. These weight values, as defined in the column "Fuel Storage Capacity per Row," represent the maximum allowable weights for ensuring that the design capacity of the fuel racks and components in the RBOF exceeds the demand from a PC-3 seismic event. The fragility values (used in the frequency analysis section of Chapter 3) are based on the minimum values as shown in the "Margin to Code" column. Incoming shipments and fuel relocation's must be verified against these data.

The fuel rack structural conditions change from the as-designed conditions during some fuel basin operating conditions, such as unlatched rack ties during fuel handling activities. This section addresses the fuel rack response and stresses for various operating conditions. Since the rack ties are active in only the north/south direction, the structural components that are significantly affected by unlatched rack ties are fuel rack vertical post I-beams, rack ties, and fuel storage hardware.

The EBS racks in the #2 storage basin were similarly evaluated for the SRS site seismic response spectrum for a PC-3 safety-class structure (Ref. 39). The scope of the calculation included the fuel rack structural components (vertical storage tubes, box beam grid members, welded connections, isolation bumpers) and the support grill over the former cask pit. The evaluation included component stresses, rack overturn stability, and rack sliding stability. The seismic analysis of the EBS racks in the RBOF for the loading of a PC-3 structure shows that the structure meets the requirements of the Uniform Building Code (UBC), Chapter 20, Division 1.

Impact of Disengaged Rack Ties

Fuel handling activities are the operating conditions that affect the fuel rack structure. During fuel handling activities, rack ties must be temporarily unlatched to allow fuel movement within the fuel rack row. Administrative and procedural controls are applied during fuel handling to limit the number of unlatched rack ties to one and also to ensure that rack ties in the fuel movement row and the two adjacent rows are properly engaged prior to and after fuel handling.

Hardware adjustments are similarly controlled in the fuel handling procedures to ensure that the normal configuration is present in the storage arrays. Occasionally, hardware is disengaged or found misaligned. Operating procedures allow for minor adjustments to restore hardware that is in this condition. Engineering and criticality safety reviews are conducted for any adjustments that could impact the spacing requirements of the NSDS.

It has been noted that movement of fuel (addition and removal) has disengaged rack ties. To address the condition of rack tie disengagement, a program has been implemented to identify the disengaged rack ties that disengage and to supplement them with a removable rack tie that will ensure engagement. The structural requirements for the modification are included in the Required Seismic Upgrade form (RSU-H-RBOF-00003), which is included in T-CLC-H-00271 (Ref. 31). The removable rack ties are designed, fabricated, and used, as necessary, to supplement disengaged rack ties.

To evaluate the operating condition of one unlatched rack tie and the occurrence of disengaged rack ties, the following 12 rack tie conditions were selected:

1. All rack ties are latched.
2. One rack tie is unlatched in the east row.
3. One rack tie is unlatched in the middle row.
4. Two rack ties are unlatched in the east row.
5. Two rack ties are unlatched: one in the east row and one in the middle row.
6. Three rack ties are unlatched: two in the east row and one in the middle row.
7. Three rack ties are unlatched: one each in the east, middle, and west rows.
8. Six rack ties are unlatched: two each in the east, middle, and west rows.

9. Seven rack ties are unlatched: two in the east row, three in the middle row, and two in the west row.
10. Seven rack ties are unlatched: two in the east row and five in the middle row.
11. Seven rack ties are unlatched: one in the east row, three in the middle row, and three in the west row.
12. Two rack ties are unlatched in the east row, with spring elements added to simulate hardware.

Discussion of Disengaged Rack Tie Analysis

The stress analysis of the rack ties showed that the rack tie assembly has the structural capacity to react the maximum rack tie loads of the disengaged rack tie conditions. The minimum capacity is produced by the support block weld of an end rack tie, and the margin to code is 5%.

To address the seismic loading on the fuel racks for the condition of an unlatched rack tie, the load that was reacted by the rack ties was applied to the parallel load path of the fuel storage hardware. The maximum load in a rack tie from the seismic analysis with all rack ties latched was applied to the fuel storage hardware in the evaluation. If the rack tie load could be reacted by fuel storage hardware, the seismic analysis with all rack ties latched is valid and the unlatched rack tie is not a significant condition. However, if the load cannot be reacted by the fuel storage hardware, additional analysis was performed to determine if the storage hardware had sufficient flexibility to deflect with the fuel rack I-beams without producing internal loads that cause overstress conditions in the storage hardware.

The disengaged rack tie analysis was conducted by releasing rack tie connections and evaluating the load redistribution. By following this process, the rack tie maximum load was increased from 645 pounds for all rack ties active (Condition 1) to 782 pounds for selected rack ties disengaged (Condition 5). However, when the maximum loaded rack tie from Condition 5 was released, the overall rack tie load reduces as shown in Condition 6.

Additional conditions were run for seven disengaged rack ties to investigate fuel rack deflection and highly loaded interior rack ties. The maximum loaded interior rack tie was produced by Condition 7. The load for this condition was used in the rack tie stress analysis to qualify the interior rack tie configuration, and the margin to code was 17%.

The deflections of the fuel rack I-beams were evaluated. The comparison of the results from the finite analysis performed showed that the stresses on the I-beam vertical posts and the rack tie structures were not limiting. The maximum absolute I-beam deflection (0.821 inches) is not large enough to cause structural contact with the adjacent structure. However, the relative displacement between I-beams (0.562 inches) was large enough to produce concerns for the fuel storage hardware. The stress analysis of the fuel storage hardware showed that the Hallam hardware for the 12-inch racks has the structural capacity to react the rack tie loads, but the MTR-9T hardware for the 9-inch racks did not have sufficient capacity. An additional run was

made (Condition 12) to address the loading on the MTR-9T hardware at the location of maximum deflection. Condition 12 was compiled by taking the condition that produced the maximum relative deflection (Condition 4) and adding spring elements, which represent storage hardware, to the location of maximum deflection. The results of this analysis showed that the storage hardware can react the load produced by the relative deflection with a positive margin to code.

The results of the disengaged rack tie analysis showed the following:

- The minimum margin to code of rack ties for the disengaged conditions is 5%.
- Rack tie loads are reduced when the maximum loaded rack tie is disengaged or fails (Conditions 5 and 6).
- Fuel storage hardware can react the rack tie load or has the flexibility to deflect with the fuel rack I-beams at the disengaged locations without exceeding the storage hardware allowable stress.
- Fuel I-beams are not overstressed.

Thus, it was concluded that disengagement or failure of all rack ties does not result in failure of the fuel racks to support the fuel.

In addition, the transfer buckets located in the RBOF disassembly basin and the large test tubes located in the inspection basin have been evaluated against the demands associated with a PC-3 evaluation basis seismic event and were found to be qualified (Ref. 145 and 146).

FUEL RACK MODIFICATIONS

Row UU - Main Storage Racks

The fuel storage hardware, PM-3A, in Row UU was found to be overstressed with a margin to code of -27%. The overstressed condition is produced by the east/west seismic loads, and failure of the storage hardware support lugs would allow the fuel to be unsupported and fall toward the east. The support lugs will not fail in the west direction because the guide plates are supported by the basin west wall when the small clearance is closed between the guide plates and the west wall. A structural modification that provides support to the fuel on the east side, at the upper elevations, will produce the structural capacity that will allow Row UU to meet the code stress limits. The structural requirements for the Row UU fuel lateral support on the east end are specified in the Required Seismic Upgrade form (RSU-H-RBOF-00002), which is included in T-CLC-H-00271 (Ref. 31). The required seismic upgrade requirements used with the Plant Modification Traveler (U-PMT-H-02122 [Ref. 42]) provide the information necessary to produce the structural modifications with the capacity to meet the seismic requirements of a PC-3 component. This rack modification has been designed, fabricated, and installed. Therefore,

Chapter 3 uses the fragility values provided for the modified configuration in support of the frequency analysis for the seismic-induced criticality events.

Bundle Storage Racks

The welds that connect the top horizontal I-beam on the southernmost beam of the frame to the vertical post I-beams were found to be overstressed. To eliminate the overstressed condition, a modification to the rack tie structure was proposed. This modification provides a parallel load path for the rack tie load and allows the rack tie support blocks and horizontal I-beams of the frames to react the load as a group. The structural requirements for this modification are specified in the Required Seismic Upgrade form (RSU-H-RBOF-00001), which is included in T-CLC-H-00271 (Ref. 31). The required seismic upgrade requirements used with the Plant Modification Traveler (U-PMT-H-02123 [Ref. 43]) will provide the structural capacity for the bundle storage racks to meet the seismic requirements of a PC-3 component. This rack modification has been designed, fabricated, and installed. Therefore, Chapter 3 uses the fragility values provided for the modified configuration in support of the frequency analysis for the seismic-induced criticality events.

2.4.2 BUILDING 245-H STRUCTURE

2.4.2.1 Controlling Dimensions for Building 245-H

The RRF (Building 245-H) is located west of the RBOF (Building 244-H). The RBOF waste cells share a common 3-foot-thick concrete wall with the RRF. Building 245-H measures 29 feet 6 inches from east to west and 40 feet from north to south. The roof is at the +17-foot elevation.

2.4.2.2 Construction Details and Materials for Building 245-H

Building 245-H (RRF) is constructed similarly to Building 244-H (RBOF). Further details associated with the construction of the RRF are available in Reference 44. A rain shelter enclosure is provided on the west side of Building 245-H to provide coverage of the truck bays and chemical pads No. 1 and No. 2. The corrugated metal rain shelter was installed in 1995 under Project #S-0117 (Ref. 45) and was designed for a wind speed of 80 mph using ASCE Standard 7-88 (Ref. 46). Four 12- by 14-foot rollup doors provide access for deionizer trailers and other delivery vehicles including chemical dumpsters. A personnel access door is also provided on the north side of the enclosure.

2.4.2.3 Building 245-H Plan and Layout

Building 245-H is a one-story building with a basement. The ground floor area is 867 ft². The building's dimensions are 34 feet wide by 25.5 feet long by 13 feet 4 inches to 18 feet high. The

construction of this building is resistant to fire. The exterior walls are 12- and 36-inch-thick concrete. The interior finish is paint in the control room. The interior wall is 36-inch-thick concrete. The wall separating Buildings 244-H and 245-H is also 36-inch-thick concrete. The floor is concrete slab on grade level. The roof is built on unprotected steel decking. The floor between the ground floor and the basement is 2-foot-thick concrete. The width and length of the basement are the same as the ground floor rooms. The construction classification according to the UBC is Type II-N.

Building 245-H consists of a resin regeneration cell, a control room, a small pump room, and a basement. The resin regeneration cell is enclosed by 36-inch-thick concrete walls. The roof is mostly unprotected metal decking with a removable wooden cover over the tanks. The cell is 12 feet wide by 28 feet long by 18 feet high.

The control room is located on the west side of the building, and it is enclosed by 12-inch-thick concrete walls. The control room is 459 ft²; its dimensions are 13.5 feet wide by 34 feet long by 13 feet 4 inches high. The room has a metal egg-crate type suspended ceiling. Ventilation ducts are above the suspended ceiling.

The pump room is on the northeast section of the building outside the boundary walls of the building. The pump room shares the concrete wall with the main building and has cement asbestos panels for the other three walls. The floor is a concrete slab; its dimensions are 11 feet 5 inches wide by 15 feet long by 7 feet 4 inches high. The pump room houses the pump connected to the nitric acid tank used in the regeneration process. The pump room also houses the stairwell that leads to the basement.

The basement is divided into three rooms. The room underneath the control room is 13 feet wide by 32 feet long by 8 feet 8 inches high. The entrance to the basement is through this room. The second room is underneath the resin regeneration cell; its dimensions are 18 feet wide by 26.5 feet long by 8 feet high. The third room is actually underneath the decontamination cell of Building 244-H. The total floor area of the basement is 852 ft². The basement is constructed of 24- and 36-inch-thick reinforced-concrete walls and roof with a concrete floor slab. The basement is a service area containing only the process piping for the regeneration operations.

RESIN REGENERATION CELL (REF. 47)

Figure 2.3-1 is a plan view of the RBOF including the RRF. The resin regeneration cell shares the 3-foot-thick wall with the RBOF waste cell. The resin regeneration cell is approximately 30 feet by 15 feet and is at the -3-foot elevation. The following three tanks mounted on concrete pads are located in this cell:

- EP57 - cation tank
- EP59 - anion resin tank
- EP63 - resin storage tank

Details on these tanks are presented in Sections 2.5 and 2.6. Access to the tanks is by means of three hatches in the roof.

BUILDING 245-H CONTROL ROOM (REF. 47)

The control room is west of the resin regeneration cell. A 3-foot-thick concrete wall provides shielding from the resin regeneration cell. The dimensions of the control room are approximately 15 feet by 29 feet at the -2-foot elevation. Monitors and controls for the regeneration process are located in the control room, including sight glasses, valve handles, radiation monitors, and other instruments described in Sections 2.5 and 2.7.

UNLOADING PLATFORM

The portable deionizers are connected to the RRF using hose box stations mounted on the unloading platform west of the control room. The platform is approximately 29 feet by 10 feet with an elevation of 2 feet 4 inches. The piping from the vessels in the resin regeneration cell terminates at the hose boxes on the platform. The resin from the portable deionizers mounted on the trailers can be transferred to the RRF by flexible piping.

PORTABLE TRAILER AND FILTER

The trailer is located west of the platform. During resin regeneration, the trailer is placed within 6 inches of the curb at hose box No. 1. Mounted on the trailer is a 100-Area portable filter and deionizer. Using the hose box, the depleted resin can be transferred to the cation tank inside the RRF. Upon completion of regeneration, the resin is transferred back into the portable deionizer. Procedures are also being established to accomplish resin regeneration for the monobed resins within the portable deionizers. The 100-Area portable filter is also cleaned at the RRF by connecting it to the hose box station and backpulsing sodium hydroxide and oxalic acid through the filter. Both of these processes are described in Section 2.5.

CHEMICAL PAD NO. 1

Chemical pad No. 1 is located north of the unloading platform. Two portable tanks (dumpsters) are mounted on the 10-foot by 15-foot 6-inch concrete pad. The nitric acid tank (EP50) is a 1,000-gallon tank, which is filled with 50% nitric acid (HNO_3). The sodium hydroxide tank (EP52) is a 1,000-gallon tank, which is filled with sodium hydroxide (NaOH). Both tanks EP50 and EP52 are connected to the system to perform resin regeneration. This connection is made by connecting the tank outlet valve to the permanently installed piping using flexible hoses. Chemicals spilled on this pad will drain to waste tank EP38. The drainage system is described in Section 2.5.

CHEMICAL PAD NO. 2

Chemical pad No. 2 is located northwest of the RRF. The target wash solution tank (EP20601) stored on this pad is a 1,000-gallon tank, which in the past stored target wash solution. (Target cleaning is no longer performed at the RBOF.) Additional chemicals stored include 1,600 gallons of 30% sodium nitrite (NaNO_2), which is stored in two tanks: a 1,600-gallon sodium nitrite tank (EP373.65) and a 1,000-gallon dumpster (503-100-1), which is used to transport 30% sodium nitrite when tank EP373.65 needs to be replenished. Chemical spills on this pad will also be pumped to waste tank EP38. The drainage system is described in Section 2.5.

2.4.2.4 Results of NPH Analysis for Building 245-H

For the NPH evaluation relative to Building 245-H, the following SSCs and corresponding NPH events were analyzed (Table 2.4-2):

- Nitric acid tank (EP50) - seismic and wind loads
- Chemical pad No. 1 berm - loads from wind-induced rain shelter failure
- Rain shelter - seismic and wind loads

Relative to the nitric acid tank and the rain shelter installed over the tank, the evaluations (Ref. 48) concluded that these components were able to withstand the seismic loads associated with PC-2 conditions.

However, for wind loads, the governing failure is the rain shelter, which was installed over the nitric acid tank. Failure of this structure was assumed to lead to failure of the tank itself. The median capacity (50% probability of failure) rain shelter is 124 mph. To reduce the consequences associated with a nitric acid tank failure from a collapse of the rain shelter, the concrete berm around chemical pad No. 1 was analyzed for impacts from framing members. In addition, the berm was analyzed for accidental impacts of a truck wheel. This is the governing case due to the magnitude of the lateral force on the berm. The analysis (Ref. 49) confirmed that the resulting loads are well within the capacity of the berm without cracking.

2.4.3 BUILDING 244-1H STRUCTURE

Building 244-1H is a supply storage and maintenance office building located about 65 feet northwest of Building 244-H.

Building 244-1H is a one-story building with a ground floor area of 1,920 ft². Its dimensions are 32 feet wide by 60 feet long by about 16 feet high. The construction of this building is noncombustible. The exterior walls are corrugated metal panels on unprotected steel frame. The construction classification according to the UBC is Type II-N.

Miscellaneous products used in the RBOF are stored in this building. Generally, the contents in the building include the following:

- Paper products
- Cleaning products
- Personal protective equipment
- Equipment parts
- Rolls of plastic and vinyl sheets

The west section of the building, about 400 ft², is a maintenance shop and offices.

There are no structures to the north wall of the building. The steam line, running parallel to the main entrance road to H Area, is north of the building. An outdoor storage area is on the south side of Building 244-1H. This yard is fenced in with a chain-link fence. The storage includes wooden pallets, noncombustible equipment, and empty 55-gallon metal drums. Chemical pad No. 1 is 45 feet southeast of Building 244-1H. Chemical pad No. 2 is 58 feet from the south wall of Building 244-1H. There is a Level D storage yard 16 feet southwest of the building. This yard is fenced in with chain-link fence. Storage is in metal containers.

2.4.4 BUILDING 244-3H STRUCTURE

Building 244-3H is a clean and used laundry storage building located about 50 feet southwest of the rain shelter.

Building 244-3H is a one-story, modular type building. This portable building is 300 ft² in area and 8 feet high. The construction of the building is partly combustible. The exterior walls are corrugated metal on wood studs. There are no interior walls. The floor is wood with vinyl tile covering. The roof is corrugated metal on wood frame. There are noncombustible ceiling tiles nailed to the interior surface of the roof. The wood frame is exposed; hence, the tiles do not create a concealed space. The construction classification according to the UBC is Type V-N.

The building is one open area used strictly for storage of clean and regulated (dirty) RBA clothing. The east end of the building is designated as an RBA for the dirty laundry.

2.4.5 BUILDING 244-4H STRUCTURE

Building 244-4H is a support building located adjacent to the south wall of Building 244-H.

Building 244-4H is a one-story, modular type building. This portable building is 240 ft² in area and about 8 feet high. The building's construction is partly combustible. The walls are corrugated metal with fiberglass insulation in the interior. The floor is wood with carpet covering. The roof is slightly pitched and is constructed of metal panels. The construction

classification according to the UBC is Type V-N. This building was moved to its present location to be in compliance with the requirements of the fire hazards analysis (Ref. 50).

Building 244-4H is a support facility, primarily used for respiratory protection items. It is not normally occupied. The building is normally kept locked when unoccupied.

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2.5 PROCESS DESCRIPTION

This section describes the individual processes within the facility. It includes details on basin process parameters, process equipment, instrumentation and control systems, and basic flow diagrams.

2.5.1 CASK RECEIPT AND UNLOADING

Casks containing spent fuel elements are delivered to the car port door by either railcar or truck trailer. The trailer is positioned inside the facility within the working limits of the 100-ton overhead crane.

2.5.1.1 Functional Requirements

Offsite shipments received at the RBOF have been certified by the Nuclear Regulatory Commission, or the Department of Transportation and the International Atomic Energy Agency. Onsite shipments of fuel are performed using offsite casks or onsite casks that are authorized for use in accordance with WSRC Procedure Manual 19Q (Ref. 51). Before the casks are received at the RBOF and placed in any of the basins, the applicable regulatory requirements mandate that the cask has been evaluated for nuclear safety. Information in the safety analysis and NSDSs for the cask and fuel are used to determine if the cask should be filled with water or steam and submerged in the basins. Each cask is also evaluated to determine if it will require steam cooling (Ref. 52). Casks are surveyed by RCO. If contamination levels on the casks are excessive, actions (e.g., decontamination) are taken before removing the casks from the vehicle.

The following nuclear safety requirements apply to cask handling activities in the basins:

- Only one shipping cask containing fuel is allowed at a time in the cask basin.
- The criticality safety of immersing a fully loaded cask must be demonstrated before placing the cask into the cask basin.

2.5.1.2 Characteristics of Cask and Fuel

The casks received at the RBOF vary in size, weight, pressure, coolant type, and internal temperature. How the casks are handled is determined in part by these parameters. Typical fuels received in the RBOF are shown in Table 2.5-1 with the cask handling requirements (as specified in the original design data) provided in Table 2.5-2. Table 2.5-2 provides the maximum height to which a cask can be lifted, which is lower than the hypothetical accident conditions required by Title 10 Code of Federal Regulations (CFR) Part 71.73 (Ref. 53).

The original design data (Ref. 7) acknowledged the hazards of handling and storing sodium bonded fuels and required (1) the detection of failed cladding before immersing the cask in water, and (2) the disposal of any exposed sodium.

Before a cask is removed from the vehicle, it is inspected for leaks and damage that may have occurred during shipment. The first step in removing the casks from the vehicle is the removal of the tie-down provisions, the sun shield, and crash shroud. These items are removed in accordance with the specific cask handling procedure. Some casks must be reoriented from a horizontal to a vertical position before being unloaded. Casks are lifted from the vehicle by the 100-ton crane using cask yokes and are placed in the cask wash pit (Ref. 28).

2.5.1.3 Washing, Sampling, and Venting

In the cask wash pit, the cask is normally washed with deionized water to remove dirt acquired in transit. The cask wash pit is equipped to perform decontamination if necessary. The cask wash pit normally drains to waste tank EP38.

The cask is also vented through the building ventilation system while in the cask wash pit to control any radioactive contamination. Before the cask can be vented, the radiation monitoring, ventilation, and air monitoring systems must be operating. If these systems are not on line, cask venting is not performed.

RCO is also present for cask venting. Appropriate personal protective equipment (respirators or air supply hoods) are worn.

Each cask type has a unique venting procedure since the inlet and outlet connections are different for each cask type. Basically, venting a cask involves positioning an 8-inch-diameter flexible hose that is connected to the building exhaust system over the cask vent outlet and opening the outlet. This process allows the pressure in the cask to be released through the exhaust system. Specific to the cask type, various types of couplings, adapters, and connectors may be used to allow the cask to vent. Typically, casks are vented for 3 minutes (Ref. 54, 55, 56).

CASK SAMPLING AND FLUSHING

If the nuclear safety analysis indicates that it is safe, the cask is filled with deionized water until water flows out of the cask vent line into the wash pit. After flushing, the drain on the cask is opened and a sample is obtained. This sample is evaluated by RCO for beta-gamma activity. If the sample activity is greater than the cask unloading limit, the cask is flushed again. The unloading limit is cask-specific and also dependent upon existing basin activity levels. The intent is to protect the basin operation limit of $7\text{E-}03 \mu\text{Ci/ml}$, thus preventing excessive contamination. This limit also ensures that the value assumed in the accident analysis ($1\text{E-}02 \mu\text{Ci/ml}$) will not be exceeded. Engineering is consulted for disposition should the cask limits identified in the procedure be exceeded. Before the cask can be placed in the cask basin, it must

be verified that placement of the cask in the basin will not cause the activity limits of the basin to be exceeded. Procedures detail the steps necessary to determine the acceptable unloading limit.

For casks with liquid coolants, the procedure for sampling is similar. Casks containing flammable antifreeze (e.g., methanol) are not allowed to be flushed to the waste tank, and special provisions are required to store the coolant until the fuel is unloaded. Once the cask is ready to be shipped back to the supplier, the coolant will be transferred back into the cask for reuse or disposed of in accordance with the applicable environmental regulations.

2.5.1.4 Desuperheating and Cask Cooling (Ref. 57)

Some dry casks may be excessively hot and require cooling using steam and/or hot water before immersion in the cask basin. This process is referred to as "desuperheating" and employs a steam desuperheater. The flow diagram for the dry cask cooling process is shown in Figure 2.5-1. This system is not currently being used.

In the desuperheating process, the hot cask is cooled progressively with steam and then hot water until the cask outlet temperature or desuperheater inlet temperature is less than 40°C. The desuperheater was designed for a maximum initial fuel temperature of 1,000°F (Ref. 57). The fuel is cooled slowly to avoid fuel damage from thermal shock.

To begin the process, the following valves and controls are set appropriately: at the desuperheater stand, at the cask cooling station, and in the Building 244-H basement. In the cask wash pit, the cask inlet is connected to the steam/water header and the cask outlet is connected to the inlet on the desuperheater. To avoid overpressurizing the cask, a pressure relief valve is provided. After verifying air flow through the cask, cooling begins.

STEAM COOLING

Steam flow to the cask is initiated by adjusting the steam control valve, while observing the desuperheater inlet temperature gage. If the desuperheater inlet temperature exceeds 115°C during cooling, deionized water flow is sprayed into the desuperheater.

If the cask outlet/desuperheater inlet temperature exceeds the dry cask station temperature by more than 50%, the steam flow is increased until the outlet temperature decreases. When the outlet temperature stops rising and does not exceed the cask station temperature by more than 50°C, cooling with hot water is started.

HOT WATER COOLING

Hot water is blended into the steam flow and the cask is cooled until the outlet temperature is less than 100°C. When this point is reached, the steam flow is discontinued and the water inlet

temperature to the cask is maintained between 95°C and 100°C by regulating the water flow to the water heater. The cask is then cooled with water until the cask exit temperature or desuperheater inlet temperature reaches 40°C. When the cask has cooled, the hoses and fittings are removed, all but two cask lid bolts are loosened, and the cask is moved to the cask basin.

2.5.2 FUEL HANDLING

During fuel handling activities, personnel work from the traveling bridge above the basins or from the basin deck area. Fuel is manipulated using underwater hand tools.

2.5.2.1 Fuel Unloading

After leaving the wash pit, the cask is lowered into the cask basin until the top of the lid is above the water surface. The remaining lid bolts are removed, and the cask is lowered to the bottom of the basin. At the bottom of the basin, the lid is removed and stored temporarily. The water above the exposed fuel provides shielding during all subsequent out-of-cask operations (Ref. 28).

Before fuel may be removed from the cask, the area radiation monitoring system in the work area must be operating, or RCO must provide constant coverage of the activity. The movement and handling of the fuel elements are planned using the appropriate procedures. The tools and materials that are necessary for handling the fuel are selected before handling begins. Examples of these tools and materials include the following:

- Unloading, transfer, and storage buckets
- Bucket inserts
- Tube bundles
- Binoculars

Based on NSDS limits for the fuel type being unloaded, the operating procedure is customized to specify the number of fuel elements to be placed in a transfer bucket. Operations, Criticality, and Engineering personnel review each customized operating procedure prior to use. The NSDS specifies the maximum safe number of elements that may be handled together (i.e., placed in a single bucket). A transfer bucket is positioned on the cask basin floor away from the cask. An unloading bucket is positioned adjacent to the cask at the level of the cask basket. Tools are used to remove a fuel assembly and to place it in the unloading bucket. The unloading bucket is then moved to a position on the cask basin floor near the transfer bucket. Unloading and transfer buckets are separated by an isolation bucket. Tools are used again to remove the assembly from the unloading bucket and to place it on a bundling table for identification. Following verification of identification, the fuel assembly is placed in the transfer bucket. Bucket inserts are required in the transfer bucket to limit the number of fuel assemblies that can be loaded into the bucket. The empty unloading bucket is then returned to the position adjacent to the cask, and the process is repeated until the designated number of assemblies have been placed in the transfer bucket. The

transfer bucket is then transported to a working or storage basin. Sometimes fuel may be bundled in the cask basin.

An RRFC (Ref. 25) has been installed in the RBOF cask basin. The intent is that a percentage of MTR type fuel assemblies will be inserted, individually, into the RRFC to determine the amount of U-235 in the fuel assembly. This measurement then becomes a verification of the fuel assembly contents provided in the manifest records accompanying the shipment.

In some situations, cask baskets are removed directly from the cask (e.g., legal weight truck cask) and are placed in basket stands on the basin floor prior to fuel unloading. Prior to removing the loaded cask basket, the basin is cleared of unsecured fuel. The cask basket is lifted and placed on the basin floor, and fuel unit unloading is accomplished as described above. Administrative restrictions generally prohibit removal from casks of cask baskets loaded with more than the safe number of fuel pieces unless specific analyses have been performed to authorize such operations.

2.5.2.2 Functional Requirements (Ref. 52, 54, 55, 56)

The following nuclear safety requirements apply to fuel handling activities in the basins:

- Cask lid bolts can not be removed nor fuel handled out of the shipping cask when the area radiation monitoring system (NIMs) are unavailable.
- The two-person rule must be observed during all fuel handling activities.
- Other than the fuel being handled, no fuel is allowed in the transfer canals or within 12 inches of any basin door adjoining the transfer canals.
- For packaging fuel, the maximum allowable number of assemblies per bundle must not be exceeded.

2.5.2.3 Fuel Bundling and Repackaging (Ref. 58)

Fuel is typically packaged in bundles in the repackaging basin. The first step is to obtain empty square tube bundles and lids from outside storage. The bundles are cleaned and assigned identification numbers. An empty bundle is placed in the bundle support frame so that the bottom rests on the bottom pan and the bundle is supported by the clips on the frame. Using the underwater hand tools, a fuel element is removed from the cask and placed in the bundle. Additional elements, as required, can be placed into the bundle as long as the NSDS limits are not exceeded. When the desired number of elements are in bundles, the top is attached to the bundle. Bundled assemblies are moved to the storage basin and placed in a rack.

In the non-EBS racks, the standard process starts with verification that the rack ties in the row where the bundle is to be stored are in the latched position. The applicable row lock is then removed after obtaining the tool from supervision, and nuclear safety controls are verified for the row identified as the storage location. Any required empty bundles necessary for spacing by the

NSDS would then be put in place or verified. The swing-down spacers and/or removable spacers are then raised/removed to allow access to the storage position. The non-EBS racks are a first-in/last-out design, and only those spacers necessary to complete the fuel bundle storage activity are handled. Should the bundle be designated for a multi-tier row, the grating is verified installed over any fuel bundles in a lower tier. Only after all applicable controls have been verified in place, the bundle is placed into the storage position. Handling is accomplished using chain hoists or similar equipment. Upon placement in the storage position, additional bundles are processed in this manner until fuel handling is completed. The row lock is then re-installed, and the rack ties are checked to ensure that they are properly latched.

In the EBS racks, the design allows for storage in essentially any available open position, and use of the first-in/last-out process is not required. The procedures and NCSEs allow for the authorized bundled fuel to be passed over stored fuel bundles to be placed into the designated storage position. There are no adjustable hardware components associated with the EBS racks.

2.5.2.4 Fuel Cropping

Preparation of fuel for bundling sometimes requires cropping of inert nonfuel-containing ends from the fuel assemblies to reduce the amount of extraneous metal.

The cropping process uses a band saw attached to a special saw table in the repackaging basin. The fuel-specific cropping dimensions are determined by Engineering, and the saw stops are set to these dimensions using a measuring rod. A second nonfuel check cut is then made, and the dimensions are checked again to ensure that the saw stops are properly adjusted. Fuel is placed on the table and held firmly against a stop on one end of the table. The saw is then used to cut the top end from the assembly. The cropped end is removed from the saw table with an underwater tool and placed in a scrap container, normally a bucket. Using a measuring rod, the saw settings are verified. If the bottom is to be cropped also, the assembly is moved to the other end of the table and this end is removed.

The saw used in cropping is electrically powered and designed for underwater operation.

2.5.2.5 Fuel Disassembly and Inspection

Fuel disassembly occurs in the disassembly basin. The basin is equipped with underwater lighting to allow adequate viewing of fuel elements. The horizontal disassembly machine is the main equipment used for fuel disassembly. The machine pulls, pushes, and rotates fuel assemblies and is used in conjunction with a tube cutter or slitter saws. A hopper is provided below the tube cutter to collect shavings or debris and to allow transfer to a suitable location for collection. However, the disassembly machine is currently not in service.

FUEL INSPECTION

Fuel inspection involves visually inspecting fuel assemblies for obvious damage as they are unloaded and verifying the fuel assembly identification.

FUEL SHIPMENT

Fuel shipment basically entails a reversal of the steps involved in fuel unloading. Fuel bundles are transferred from the storage basin via canals to the cask basin where cask loading takes place. The cask lid is put in place and the cask is lifted to the surface where the lid bolts are tightened. The cask is transferred to the cask wash pit where it is surveyed and decontaminated as necessary. Finally, the cask is transferred to either a railcar or truck.

2.5.3 BASIN WATER SYSTEM

2.5.3.1 Functional Requirements

The function of the basin water system is to provide adequate shielding from the fuel elements stored in the basins during normal and accident conditions. According to the original design data (Ref. 7) and under normal conditions, a dose of 0.1 mrem/hr would not be exceeded. A secondary function of the basin water system is to contain any fission products that may escape from the fuel matrix and/or fuel cans/bundles during fuel storage or under accident conditions (Ref. 59). In addition, proper chemistry of the basin water minimizes the potential of fuel failing and releasing fission products.

The RBOF consists of six basins that contain approximately 500,000 gallons of deionized water. Fuel is normally stored in the two storage basins but may be placed in the other basins. The basins are connected via a canal system using underwater doors. The basins are connected such that the water level is uniform; the current hazards/accident analysis does not assume that the basins are isolated through the use of the canal doors.

The water level in the basins is necessary for protecting the facility workers and the public in the event of a criticality (Ref. 23). The water level provides attenuation of the direct radiation dose to the facility worker. In addition, the basin water level minimizes the release (to the air above the water surface) of certain nonvolatile fission products. This "scrubbing" effect occurs after a postulated fuel melt from a process- or NPH-induced criticality event. The contamination and turbidity levels of the water in the basins must also be controlled to allow safe underwater fuel handling activities and to control contamination levels to as low as reasonably achievable. The systems for maintaining the basin water are described below.

2.5.3.2 Basin Water Level Control

The water level in the basins is required to be at least -18 inches with respect to the Building 244-H floor grade (Ref. 23). During normal operation of the system, all basins are filled to the required level, which corresponds to a nominal level of -10 to -12 inches below the operator floor. If the water level drops to -14 inches, a low-level alarm is sounded. Also, if the water level drops to -17 inches, a low-low interlock shuts down the basin water discharge pumps. The water supply to the basins is either water from the power house or water that is recycled through the filter deionizer loop. The source of water to the basins is selected by setting the three-way inlet valve for either recycle or fresh water. The recycle loop is described in Section 2.5.3.3.

The process water systems are discussed further in Section 2.8.2. Each basin includes a pneumatic dip tube for measuring the basin water level and a submerged centrifugal pump for removing water from the basins. The water level in each basin is established by the setpoint on the level controller provided in the control room. The level controller regulates the water input to the basins. When the level in a basin falls below the setpoint, the controller opens the water control valve allowing more water into the basin. When the level reaches the setpoint, the controller closes the water control valve. Input to the controller comes from a dip tube mounted in the basin (Ref. 60).

Water inlet to the basin is capable of being controlled automatically, or additions are made under direct operator control. However, the water outlet, and thus lowering of the basin level, is not subject to automatic control. Water is removed from the basins by a submerged centrifugal pump in each basin. The operation of the pumps is not controlled automatically. To lower the basin water level, the discharge pump must be manually operated from the valve stand. The water level is monitored by operator rounds using the basin water level ruler located in the storage basin. There is a high- and a low-level pressure switch alarm for each basin. A high level in a basin will close the inlet valve to that basin as well as the main deionized water inlet valve. Table 2.5-3 provides the basin level alarm setpoints.

The discharge pumps are interlocked with low-low-level pressure switches (not level controllers) for each basin so that all pumps will be de-energized if the water level falls below the low-low-level setpoint in any basin. In addition, float switches are provided on each of the pump's suction piping to open a suction vent valve once the water level decreases below the -36-inch level. Table 2.5-4 provides the basin pump setpoints. With these setpoints and given the remote chance of the filter deionizer system leaking while the basin pumps continue to operate, a small percentage of the current fuels stored in the RBOF could become uncovered (Ref. 61). However, the length of fuel uncovered would not lead to a significant heat transfer loss to such an extent so as to compromise fuel cladding integrity.

Separating the inlet and discharge controls is necessary for proper waste handling and water purification. To raise the level in the basins, the level controller need only be set to a higher setpoint. To lower the level of the water, the setpoint must be lowered and the discharge pumps must be manually started and valved to either the recycle loop or waste tank EP38. The fact that

two actions are necessary for lowering the basin water level also protects against inadvertently pumping a large amount of water out of the basins.

2.5.3.3 Basin Water Purification

According to the original design (Ref. 59), three degrees of water contamination (innocuous, nocuous, and abnormally nocuous) were provided with three types of water disposal systems as follows:

- Innocuous Contamination - Flush to Seepage Basin

Innocuous contamination was defined as deionized water with fission product concentration less than $5\text{E-}04 \mu\text{Ci/ml}$. To maintain clarity and low activity, a flow equivalent to 1/5 basin volume turnover per day was maintained through basins in this condition. Water was to be discarded to the seepage basins with a continuous makeup of fresh deionized water. Due to the retirement of the seepage basin, this option is no longer available.

- Nocuous Contamination - Recirculate to Decontamination Equipment

Nocuous contamination was defined as deionized basin water with fission product concentration less than $5\text{E-}02$ and greater than $5\text{E-}04 \mu\text{Ci/ml}$. Deionized water makeup was to be provided for losses from the normally closed system.

The decontamination equipment was to provide collection and disposal of radioactivity in the form of dissolved and suspended solids in the waters from any of the basins at a maximum total flow of 75 gpm and a minimum of 20 gpm from any one basin.

Collection of radioactivity by filtration and deionization of basin water at a maximum rate of 75 gpm would process the entire volume of all basins approximately once every 4.5 days. Water would be pumped from the basins through a backwash filter and a mixed bed exchanger and returned to the basins. Flows from the basins would be intermixed, except when handling only a single basin.

To permit decontamination of basins at flows greater than 75 gpm, pipe headers from the basins and return were sized for 150-gpm flow. Space and blanked flanges are provided for future installation of required piping and equipment.

Particulate solids 1 micron and larger (normal concentration less than 10 ppm) will be collected in the filter on Porostone filter elements, precoated with a filter aid.

Dissolved solids (normally less than 1 ppm) in the filter discharge stream will be removed during flow through a mixed bed exchanger.

- Abnormally Nocuous Contamination - Flush to Waste Tank

Abnormally nocuous contamination is defined as basin water with fission product concentration greater than $5\text{E-}02 \mu\text{Ci/ml}$. Such basin water would be pumped directly to waste tank EP38.

The conductivity of the water is typically maintained at less than $3.0 \mu\text{mhos/cm}$ with a maximum of $10 \mu\text{mhos/cm}$. The radioactivity level of the water typically ranges from 0.5 to $1.5\text{E-}04 \mu\text{Ci/ml}$. The normal inventory of 500,000 gallons of basin water is thus 0.1 to 0.3 Ci (Ref. 28). Limits and requirements for the basin water system are shown in Table 2.5-5 (Ref. 62).

Visual inspections of the RBOF basins and fuels have been conducted to determine if there were signs of visible corrosion products on the fuels themselves or aluminum bundles or cans (Ref. 63). From this inspection, no signs of visible corrosion were noted on the aluminum-clad NEREIDE fuel (oldest aluminum-clad fuel stored in the RBOF), the aluminum can which housed this fuel, any visible storage tubes, or the aluminum storage racks. This inspection determined that the above components showed no signs of pitting corrosion at times up to 30 years of storage, which was attributed to the low conductivity of the basin water and the corresponding low impurity content of corrosive ions.

RECYCLE LOOP SYSTEM

The filter deionizer recycle loop system is used to decontaminate water in the basins and canals. The system consists of a booster pump, precoated Porostone filter, deionizer vessel, afterfilter unit, water surge tank, precoat supply tank, individual basin pumps, and admix supply tank. Figure 2.5-2 shows the flow diagram for the system.

The basin water system normally operates in the recycle mode. All basins are filled to the required levels, and the three-way valves on the inlets are set on recycle. Normally, the discharge pump on the most contaminated basin is discharging to the filter deionizer system. Normally, two basins are simultaneously connected to the filter deionizer, but the capability exists to align only one basin to the deionizer (Ref. 28). The total flow through the system is limited by the filter pressure drop, which is limited to 25 psig (Ref. 64).

The recycle loop booster pump (EP13) provides the head necessary to move the water through the filter deionizer system. This pump normally operates at 18 to 22 psig (Ref. 64). Water flows from the booster pump to Porostone filter (EP14). The filter contains tubular type elements, which remove particles 1 micron in diameter and larger, reducing the turbidity of the water.

From the filter, the water flows into the deionizer (EP15). The deionizer contains 24 ft^3 of cation resin and 36 ft^3 of anion exchange resin mixed together (Ref. 60). The deionizer typically reduces the conductivity from about $2.5 \mu\text{mhos/ml}$ to less than $1 \mu\text{mhos/ml}$. The deionizer

normally operates at a flow rate of 75 to 125 gpm. The flow rate is controlled by the valve positions. The afterfilter (EP15.3) collects particles that may escape through the deionizer.

Before flowing into the recycle loop, the water flows through a conductivity cell in the basin area. The conductivity is read from a panel in the control room. From the conductivity cell, the recycled water flows to the recycle loop surge tank (EP48). This tank serves as the head tank from the loop and prevents surges in the filter deionizer loop. The tank is located on the roof of Building 244-H. Recycled water from tank EP48 flows back into the basins and canals.

PRECOAT PROCESS

Before the recycle loop is operated, the Porostone filter must be precoated. The filter is coated with a 3% slurry of Celite precoat material. The precoat protects the filter by preventing solids from becoming imbedded in pores of the tubes.

The precoat material is prepared in the precoat tank (EP17). This tank is located in the chemical preparation area of the control room. The precoat operation is started by adding deionized water to the tank and starting the agitator. Fifteen pounds of Celite 545 precoat material is then added to the tank, and the mixture is agitated for 5 minutes (Ref. 65).

COATING THE FILTER

To coat the filter, the precoat tank and filter are valved out of the recycle system. All basin recycle pumps are stopped. Precoat solution is pumped using pump EP18 through the filter and back to the precoat tank. Once precoating has begun, the flow continues uninterrupted for approximately 20 minutes to prevent precoat material from falling off the filter. When the precoat process is complete, flow through the filter is switched back to water from the basin being decontaminated.

FILTER BACKWASHING (REF. 65)

When the pressure drop becomes excessive ($\Delta P > 25$ psig), the filter cake must be removed from the filter. This process is referred to as "backwashing."

To backwash the filter, the filter is first drained to waste tank EP38. The filter dome is pressurized with air to 100 psig. The pressure is then released rapidly, allowing the filter cake to drain to waste tank EP38. The filter dome is next pressurized with water. The pressure is again released rapidly with the filter cake waste draining to waste tank EP38. Additional backflushes of the filter dome are performed as required to clean the filter.

FLUSHING THE FILTER (REF. 66)

The filter is also periodically cleaned chemically with caustic and oxalic acid. To clean the filter, the deionizer loop is shut down and any resin in the deionizer is transferred to the RRF. Oxalic acid solution is prepared in precoat tank EP17 by adding oxalic acid crystals to water and agitating the tank. A 4% to 6% solution is used.

After draining the filter dome to waste tank EP38, the oxalic acid solution is circulated through the filter loop for approximately 3 hours. The oxalic acid solution is drained to waste tank EP38 after the solution has been sampled. The filter loop is flushed with deionized water and drained to waste tank EP38. A caustic solution (approximately 30%) is prepared in precoat tank EP17 by adding 50% caustic to water. The caustic is circulated in the filter loop in a process similar to the circulation of oxalic acid.

2.5.3.4 Process Equipment (Ref. 60)

BASIN AND CANAL PUMPS (EP7.1 THROUGH EP7.8) (REF. 67)

The basin and canal pumps are "chempump" canned centrifugal pumps designed for submerged operations. These single-stage pumps are furnished with the following:

- Carbon bearings
- A motor
- A high-pressure electrical lead connector
- An extended conduit of schedule 40 stainless steel pipe

The pumps contain a 5-inch-diameter impeller.

All pumps, except the disassembly basin pumps, have a self-cleaning discharge filter of 80 mesh screen and a circulation tube to cool the motor section and lubricate bearings. The disassembly basin pump is supplied with deionized water to cool the motor section and to lubricate the bearings. Pumps are driven by totally enclosed motors (3 phase, 60 Hz, 3,450 rpm, 440 volt) with Type H insulation (Table 2.5-4).

CASK WASH PIT PUMP (EP6) (REF. 68)

The cask wash pit pump in the control room basement circulates or purges cask coolant and pumps liquid out of the wash pit. The unit is a Durco Series HR, single-stage, horizontal, self-priming, stainless steel centrifugal pump. The casing and cover are Type 304 stainless steel. A 9-inch Type 304 stainless steel impeller is used. A ceramic-to-graphite mechanical seal with Type 304 stainless steel retainers and springs is used on the pump shaft. Carbon throttle

bushings are used, and the stuffing box is fitted with a water inlet for flushing the seal and bushing. The motor for this pump is a 7-1/2-hp, 440-volt, 60-cycle, 3-phase motor.

RECYCLE LOOP BOOSTER PUMP (EP13) (REF. 67)

This pump, which is located in the control room basement, develops the head required to pass at least 75 gpm through the filter deionizer system. The pump suction is supplied by the submerged pumps of the basin and canal. This pump is a canned centrifugal "chempump" similar to the basin pumps but does not operate submerged in water. Filter aid is fed to the stream on the discharge side of the pump. The pump has a 7.5-inch-diameter impeller to boost system pressure from 25 psig to about 80 psig. Pump material wetted by the process stream is Type 316 stainless steel, and the pump is mounted on a carbon steel bracket and base plate. The pump does not have lifting rings or extended electrical conduits. Bearing flush water on this pump is supplied from the discharge.

The motor section of this pump is filled with oil to dissipate heat.

FILTER PRECOAT PUMP (EP18) (REF. 68)

The filter precoat pump in the chemical preparation area of the control room pumps a 3% filter precoat slurry to the recycle loop filter. The unit is a Durco Series H1, single-stage, horizontal, stainless steel centrifugal pump. A 7-inch Type 304 stainless steel impeller is inserted in the Type 304 stainless steel casing and end cover designed for an impeller up to 7.5 inches in diameter. A mechanical seal used on the pump shaft has an external water flushing connection.

The motor driving this pump is a 3-hp, 440-volt, 60-Hz, 3-phase motor.

FILTER AID PUMP (EP24) (REF. 69)

This pump, which is mounted on the admix tank in the control room, pumps a slurry of filter aid to the filter deionizer loop until the filter aid is deposited on the filter. It is a Model 601 Precision Chemical diaphragm type pump, which accurately meters slurries in concentrations to 5%. A shaded pole motor and gear train in the oil-filled case operate a cam to flex the diaphragm in the pump head. Output can be adjusted up to 2.5 gpm by setting a pointer on top of the unit. Maximum pressure developed is 125 psig. A solenoid valve and timer arrangement provides a 6-second deionized water flow of the pumping chamber every minute. Bearing cooling water is provided from the demineralized water system.

The integral motor operates on 110-volt, 60-Hz current.

PRECOAT TANK (EP17) (REF. 70)

The precoat tank in the chemical preparation area of the control room is used to make up and apply zeolite precoat to the loop filter. The tank is fabricated of 16-gage Type 304 commercial grade stainless steel and is 3 feet in diameter by 3 feet high with a dished bottom. Capacity is about 160 gallons. A split-hinged cover permits one-half of the cover to open. Two 3-inch nozzles on the fixed section of the cover and a 3-inch bottom discharge nozzle have carbon steel lap joint flanges.

A model PH3 Chemineer portable agitator is inserted through an opening in the tank top and is supported by a flange mounted on the tank side. It has a totally enclosed motor (440 volt, 3 phase, 1/3 hp, 1,750 rpm) with gear drive for 430-rpm output. The agitator shaft and propeller are Type 304 stainless steel.

ADMIX TANK (EP23) (REF. 70)

The admix tank in the chemical preparation area of the control room is used to prepare 3% to 5% slurries of filter aid. The tank is identical to the precoat tank (EP17) except for the size and design of the nozzles. The admix tank has three stainless steel nozzles: 1- and 2-inch nozzles on the fixed section of the cover and a 1-inch bottom drain nozzle. The agitator is identical to that on the precoat tank. Capacity is about 160 gallons.

RECYCLE LOOP FILTER (EP14) (REF. 71)

This filter is located in the decontamination cell. The unit removes solids 1 micron and larger at flow rates of 75 to 125 gpm with a pressure drop of up to 25 psig. The filter body and tube sheet are Type 304 stainless steel. Design pressure is 125 psig.

Twenty-seven Porostone filter tubes, 36 inches long (two 18-inch joined pieces), are arranged in the tube sheet to allow 1.5 inches between tube surfaces and provide a filter area of 91 ft². The tube sheet and tubes are removable as a unit.

Vessel height is 78-5/8 inches. There is a 2-inch vent-air blow nozzle, a 3-inch outlet nozzle, a 3-inch inlet nozzle (all side-entering), and a 3-inch bottom outlet nozzle. All nozzles terminate facing downward and in the same plane to limit the amount of fixed exposed piping within the decontamination cell. The vessel bottom slopes to facilitate solids removal during the backwash operation.

DEIONIZER AFTERFILTER (EP15.3) (REF. 72)

The afterfilter in the control room basement removes resin particles that break through the deionizer unit. The unit is a Cuno Type EM Auto Clean filter containing three cartridges and is 2.25 inches in diameter by 8 inches long with 0.0035-inch spacing between leaves. All material,

including elements, is Type 304 stainless steel. The filter is designed for a 125-psi operating pressure. The inlet and outlet connections are 3 inches, and the bottom drain is 2 inches. The filter leaves are cleaned by rotating a handle on the top of the filter.

DEIONIZER (EP15) (REF. 73)

The deionizer vessel in the decontamination cell removes radioactive and nonradioactive ions from the water flowing through it. All wetted parts are Type 304 commercial grade stainless steel. The vessel contains 60 ft³ of mixed resins. It is 5 feet in diameter by 5 feet high with a dished top and bottom. Design pressure is 150 psi. The vessel has carbon steel lifting lugs and legs. Total height, including legs and external piping, is about 11.5 feet.

An 18-inch manhole on top of the vessel has a carbon steel closure flange. Five nozzles terminate pointing downward in the same plane to limit the amount of fixed exposed piping in the decontamination cell. These nozzles include the following:

- A 2-inch vent nozzle
- A 3-inch water inlet nozzle
- A 2-inch resin inlet nozzle
- A 2-inch resin discharge nozzle (all top- or side-entering)
- A 3-inch water outlet nozzle on the vessel bottom

The description of the basin water system contained in this section is based on the references given and the drawings shown in Table 2.5-6.

DEIONIZED WATER HEAD TANK (EP49) (REF. 70)

This 160-gallon vessel located on the Building 244-H roof is a vent point in the deionizer water supply from the power house (Figure 2.5-3). It acts as a siphon break to prevent water transfers from the basins in case of a line break or low pressure upstream. The unit is fabricated of Type 304 commercial grade stainless steel. It is a vertical cylinder, measuring 3 feet in diameter by 3 feet high, with a dished bottom. A cover bolted to the tank top has 1-, 3-, and 4-inch nozzles. Water enters through the 3-inch top nozzle at the center of the bottom. An inverted "J" on the 1-inch top nozzle vents the tank to the atmosphere. A float type level controller is mounted on the 4-inch nozzle. The 3- and 4-inch nozzles on the top and the 3-inch bottom nozzle have carbon steel lap joint flanges. The vessel is supported on carbon steel pipe legs.

RECYCLE TANK (EP48) (REF. 74)

The recycle tank is also mounted on the roof. It acts as a head tank to prevent surges in the filter deionizer loop flow. The vessel is a horizontal cylinder, measuring 5 feet in outside diameter by

12 feet long, with dished heads. The vessel is fabricated from 3/16-inch-thick Type 304 commercial grade stainless steel. Design pressure is atmospheric. A 1-inch dip tube extends down to 2 inches above the tank bottom to measure liquid level. Tank capacity is about 1,900 gallons. The vessel rests on carbon steel saddle supports.

PROCESS INSTRUMENTATION AND CONTROLS

Basin Level Controls

The water level in the RBOF basins is controlled by pneumatic level controllers mounted on panels in the Building 244-H Control Room. The setpoints on the controllers are normally set to maintain the water level at least -10 to -12 inches with respect to the floor grade. Each basin is provided with independent pneumatic controllers. The normal range of operation of the controller is over a range of 13 inches of water.

The level of water in the basins is measured using a dip tube. The dip tube generates a pneumatic signal proportional to the basin level and sends it to the controller, which compares the level to the setpoint. If the level is below the setpoint, the controller sends a pneumatic signal to a solenoid valve. The solenoid valve opens the inlet water control valve on the basin to allow water in from the water supply. The level controller does not regulate the water outlet from the basins. To lower the basin water level, the discharge pump for the basin must be manually started and the controller must be set at a low setpoint.

Interlocks

The interlock relays for the basin pumps are in a cabinet (Panel 480.2-9) located at the south end of the Building 244-H Control Room. Each basin contains a low-low-level pressure switch that senses the dip tube level. The pressure switch will open when the basin level is below a certain level. Table 2.5-4 provides the low-low-level setpoints. The opening switch de-energizes a 120-volt relay, which opens the circuit for the booster pump motor (EP13) as well as all individual discharge pumps.

In addition, float switches are provided on each of the pump's suction piping to open a suction vent valve once the water level decreases below a certain level. Table 2.5-4 provides the water level setpoints.

The system is also interlocked such that a high level in any basin closes the inlet valve for that basin and the main deionized water inlet valve.

2.5.4 RESIN REGENERATION PROCESS

2.5.4.1 Functional Requirements

The resin regeneration process is carried out at Building 245-H. The function of the resin regeneration process is to regenerate anion and cation resins used in the water purification system of the RBOF and 100-Area deionizers. Anion exchange resin is regenerated by contacting the resin with dilute sodium hydroxide, and cation exchange resin is regenerated by contact with dilute nitric acid.

2.5.4.2 Process Flow

REGENERATING RESIN FROM THE RBOF

The resin regeneration process flow diagram is shown in Figure 2.5-4. Resin from deionizer EP15 is transferred by a combined flow of air and water. The resin is received through permanently installed piping in cation tank EP57. Before separating anion and cation resin, the resin is depleted by adding 50% sodium hydroxide to cation tank EP57. The anion resin is separated and, by an upflow of water at 60 gpm and air at 60 cfm, sent to anion tank EP59 (Ref. 75).

Cation resin in cation tank EP57 is regenerated by adding a nitric acid solution from portable acid tank EP50. Acid is pumped by acid metering pumps EP51A and B. Anion resin is regenerated in anion tank EP59 by adding a caustic solution from portable caustic tank EP52. Caustic is pumped by caustic metering pumps EP53A and B. The precise amount of resin that is regenerated and waste formed in each step of the process is regulated by controlling the acid and caustic flow rates from the pumps. Cation resin is depleted at a rate that ranges from 1.1 to 1.6 gpm. The depletion rate for anion resin ranges from 0.7 to 1.1 gpm (Ref. 75). The rate is determined by the time settings on the metering pumps and by comparing the levels in the acid and caustic tanks before and after running the pumps. After regeneration, the resins in both tanks EP57 and EP59 are rinsed with deionized water. The resin can then be transferred back to deionizer EP15 in Building 244-H.

REGENERATING 100-AREA RESIN

The resin from portable 100-Area deionizers is regenerated in a similar process (Figure 2.5-4). The portable deionizers may contain a maximum of 36 ft³ of anion resin and 24 ft³ of cation resin. The resin is regenerated using the hose box stations on the west side of Building 245-H. The deionizers are mounted on a trailer adjacent to hose box No. 1. Using flexible hoses, the deionizer resin outlet is connected to the hose box terminations. Using compressed air and water, the resin can be transferred to cation tank EP57 in Building 245-H. In cation tank EP57,

resin from the portable deionizer is not required to be depleted. After regeneration, the resin is transferred back to the portable deionizer.

Also, monobed resin can be regenerated while in the portable deionizer without transferring the resin into cation tank EP57. Using flexible hoses, connections are made at hose box No. 1 to the deionizer inlets and outlets, and chemicals are transferred into the portable deionizer to regenerate the resin.

2.5.4.3 RRF Process Equipment

HOSE BOXES

The hose box stations located west of the RRF provide the means for connecting portable deionizers to the RRF. The hose boxes are also used to transfer sodium nitrite, caustic, nitric acid, and phosphoric acid to the RRF and the RBOF waste and decontamination cells. The hose box layout is shown in Figures 2.5-4 and 2.5-5.

The hose boxes are fabricated of 16-gage stainless steel. Each hose box contains various valves, couplings, and connectors used to connect flexible corrugated stainless steel tubing. Typically, one end of the flexible connector is flanged while the other end has a coupling or adapter.

Portable deionizers are usually connected at hose box No. 1. This hose box has terminations for connecting compressed air, waste, resin inlet, resin outlet, caustic, nitric acid, and deionized water.

Hose box No. 2 is used in transferring sodium nitrite, caustic, nitric acid, and phosphoric acid. The sodium nitrite and target wash pumps are mounted in this hose box. The sodium nitrite tank is situated just outside of the hose box to the north. Since target washing activities are not performed presently, the target wash pump is not operated.

Both hose boxes are provided with drains that slope to waste tank EP38.

CATION TANK (EP57)

This 900-gallon cylindrical tank is fabricated of 3/16-inch-thick Type 304 stainless steel (Figure 2.5-4). The vessel has a diameter of 4 feet and is 9 feet high. It contains five sight glasses for visual inspection of the tank level. It has a dished top and bottom. This tank receives mixed ion exchange resin (Ref. 28).

ANION TANK (EP59)

This 500-gallon cylindrical tank is fabricated of 3/16-inch-thick Type 304 stainless steel (Figure 2.5-4). The vessel has a diameter of 3 feet and is 9 feet high. It contains four sight glasses. This tank receives anion exchange resin (Ref. 28).

STORAGE TANK (EP63)

This 900-gallon cylindrical tank is fabricated of 3/16-inch-thick Type 304 stainless steel. The vessel has a diameter of 5 feet and is 5 feet high. It contains three sight glasses. This tank is used to store spent resin until it can be packaged for burial (Ref. 28).

ACID STORAGE TANK (EP50) (REF. 76)

This 1,000-gallon portable tank contains 50% nitric acid. It is located on chemical pad No. 1. This tank is moved to Building 211-H to be filled, and it is equipped with an air vent line, dip tube, and filter regulator (Ref. 28).

CAUSTIC STORAGE TANK (EP52) (REF. 77)

This 1,000-gallon portable tank, located on chemical pad No. 1, contains a 50% caustic solution. To prevent freezing, the tank can be heated by 15-psig steam supplied from the steam header. A temperature element mounted in caustic storage tank EP52 is used to regulate the steam supply. The tank is moved to Building 211-H to be filled, and it is equipped with an air vent line, dip tube and filter regulator (Ref. 28).

SODIUM NITRITE HOLD TANK (EP373.65)

This 1,600-gallon tank is located on the east side of hose box No. 2. Sodium nitrite tank EP373.65 receives sodium nitrite from dumpster 503-100-1. This unit is fabricated of 3/16-inch-thick Type 304 stainless steel. The tank receives 30% sodium nitrite mixed with deionized water to yield a 5% solution (Ref. 28) (Figure 2.5-6).

SODIUM NITRITE TANK (EP503-100-1)

This 1,000-gallon portable tank is used to transport 30% sodium nitrite. The tank is located on chemical pad No. 2, adjacent to tank EP373.65. The tank is moved to Building 211-H to be filled, and it is equipped with a filter regulator (Ref. 28).

TARGET WASH SOLUTION TANK (EP20601)

This 1,000-gallon tank is located on the northeast side of the hose box No. 2. The tank is abandoned in place and is not used. It previously contained a solution that was used in the target washing process. This vessel is fabricated of 3/16-inch-thick Type 304 stainless steel. The target wash solution consisted of 1% sodium dichromate, 10% phosphoric acid, and deionized water (Ref. 28) (Figure 2.5-6).

PUMPS (REF. 28)

Acid Pumps (EP51A and B) (Ref. 78)

The acid pumps (EP51A and B) are metered pumps, each capable of delivering 54 gal/hr at 100% of its setting. Each pump is driven by a 3/4-hp electric motor started in the Building 245-H Control Room. Pumps EP51A and B are operated simultaneously and in parallel. (Figure 2.5-5).

Caustic Pumps (EP53A and B) (Ref. 78)

The caustic pumps (EP53A and B) are metered pumps, each capable of delivering 59 gal/hr at 100% of its setting. Each pump is driven by a 3/4-hp electric motor started in the Building 245-H Control Room. Pumps EP53A and B are operated simultaneously and in parallel.

Sodium Nitrite Pump

This is a Jabsco pump that transfers a 30% sodium nitrite solution to the sodium nitrite tank. This pump is mounted in the hose box.

RRF INSTRUMENTATION AND CONTROLS

The resin regeneration process is a manually controlled process. Valves are opened by manually turning valve handles in the control room. The level of process tanks is determined visually through sight glasses. The rate of resin regeneration is controlled by regulating the flow of deionized water and acid/caustic to the resin regeneration cell. Resin transfers are performed using compressed air from compressor EP68 and deionized water. These transfers are done by manual regulation of the compressor output pressure. Radiological monitors and instrumentation are described in Section 2.7.

2.5.5 RBOF/RRF VENTILATION SYSTEMS

2.5.5.1 Functional Requirements

The RBOF/RRF ventilation systems minimize the airborne radioactivity both inside and outside the facility through the use of a "once-through" air flow system and filters on the building exhaust. Because of the once-through ventilation system, air activity levels do not tend to increase with time. The filters remove contamination that would otherwise be released to the atmosphere. Also, the facility is maintained at less than atmospheric pressure allowing all air to be filtered before discharge. The basin areas are at a lower pressure than the rest of the building, and a separate ventilation system supplies air to the control room, offices, and change room. A separate vessel vent system is also provided for the process vessels in the waste cell and resin regeneration cell. This system is similar to the building system but uses acid-resistant components (Ref. 7, 22). Confinement aspects of the ventilation systems are discussed in Section 2.6. Additional details relative to the original design can be found in DPWA-61-212 (Ref. 7).

2.5.5.2 Air Flow and Distribution

The RBOF/RRF ventilation systems consist of the following:

- Several filtered and conditioned supply fan systems
- One filtered primary building exhaust system
- A louvered wall exhaust from the electrical room
- A roof exhauster from the change rooms
- A filtered vessel vent offgas system

The air flow diagram for the RBOF and RRF is shown in Figures 2.5-7 and 2.5-8. The air supply fans and motors for the RBOF and RRF are listed in Table 2.5-7.

Air flow in the facility is from areas of lower contamination to areas of higher contamination. The air supply to the offices, control room, and change room area is separate from the building supply since contamination is not expected in this area. The exhaust air from the control room flows down through the maintenance shop and stairway into the basement before finally being exhausted. Doors and louvers are provided to ensure flow in the desired direction and to minimize reverse flow (Ref. 7).

EXHAUST SYSTEM

The primary building exhaust system discharges approximately 23,750 cfm via a stack measuring 5 feet in diameter by 53 feet high. The electrical switchgear room discharges 400 cfm through an

unmonitored, unfiltered wall louver at approximately 12 feet above ground. The change room (restroom) area discharges 1,100 cfm via an unmonitored, unfiltered roof fan. The vessel vent system discharges approximately 510 cfm through a 10-inch-diameter stack. The exhaust fans and motors are listed in Table 2.5-8. Supply and exhaust fans do not have backup power.

All air exhausted from the basins, control room, and offices is filtered by a filter bank on the roof that consists of prefilters followed by a filter bank (EP908.2). The main exhaust fan is mounted adjacent to the filter bank. The exhaust is sampled and monitored for beta-gamma contamination before being discharged to the stack.

VESSEL VENT SYSTEM

The vessel vent system filters receive exhaust from process vessels in the RRF. The RBOF waste cell and the hose box stations are piped to waste tank EP38. Waste tanks EP38 and EP37 exhaust through the vessel vent system. This system provides the primary confinement for gaseous and airborne effluents from these process vessels. The exhaust air from these areas flows into a stainless steel plenum, which goes up to the roof. The exhaust is filtered by a prefilter and particulate filter before exiting via a 10-inch-diameter pipe. Exhaust from the vessel vent system is sampled and monitored for beta-gamma contamination. The stack monitoring systems are described in detail in Section 2.7.

2.5.5.3 Equipment and Controls

INSTRUMENTATION AND CONTROLS

The use of instrumentation in the ventilation system is limited to two interlocks on the main exhaust fan. The main exhaust fan is interlocked with the NIMs to shut down when the NIMs are activated to prevent the spread of contamination outside of the facility during a nuclear incident.

The other interlock is between the main exhaust fan and the supply fans. A pneumatic pressure switch set at 2 inches water column is mounted in the main exhaust suction duct. The pressure switch prevents the supply fans from operating when the exhaust fan is not running, which protects against pressurizing the facility and allowing airborne contamination to flow unfiltered from the facility.

2.5.6 WASTE HANDLING

The purpose of the waste handling process is to collect waste generated in the facility and to prepare the waste for transfer to those facilities that treat or dispose of the waste.

Process wastes that are handled in the RBOF and RRF include solid waste generated from fuel handling activities and resin regeneration. Liquid waste in the facility has many sources. This

section describes the systems and processes for treating and disposing of waste in the RBOF and RRF.

2.5.6.1 Liquid Waste Handling (Figures 2.5-2, 2.5-8, and 2.5-9)

FUNCTIONAL REQUIREMENTS (REF. 79)

All process liquid waste from the facility is collected in waste tank EP38 for transfer either to the GP evaporator (feed tank 717) or to a waste tank in the H-Area tank farm. Previously, low-level waste went to waste tank EP37 for eventual discharge to the seepage basins. These basins have since been closed. Waste tank EP37 now serves as a surge tank and collection tank for some floor and sump drains. Waste tank EP37 is connected via an overflow line to waste tank EP38.

Under normal circumstances, waste batches from waste tank EP38 that do not meet the Waste Acceptance Criteria (WAC) are not sent to the GP evaporator (feed tank 717). This waste is adjusted to meet the WAC and is sent to tank 21 or 23 in the tank farm.

In transferring waste to waste tank EP38, a 75% level in the tank is not exceeded. This limit is verified by monitoring the level of the tank in the control room during any waste transfer (Ref. 79).

WASTE SOURCES

The sources of waste received in waste tank EP38 and sent to the GP evaporator (feed tank 717) or the tank farm are listed below (Ref. 79):

<u>Waste Type</u>	<u>Destination</u>
Filter backflush	Tank 21-H/23-H
Cask flush	Tank 21-H/23-H
Caustic flush	Tank 23-H
Waste cell sump	GP evaporator (feed tank 717) or Tank 21-H/23-H
Deionizer waste	GP evaporator (feed tank 717) or Tank 21-H/23-H
Filter oxalic acid flush	Tank 23-H
Basin/canal water	GP evaporator (feed tank 717) or Tank 21-H/23-H

WASTE FROM BASIN/CANAL WATER

Basin/canal water is normally recycled through the filter deionizer loop. Water from the basins can be transferred to waste tank EP38 in 4,000-gallon batches (Ref. 79). The transfer can be

performed while the filter deionizer system is on line. Before the transfer, the level of waste tank EP38 is determined, and the tank is emptied if the level is greater than 25% of capacity (nominal capacity is 6,400 gallons). To send the water to waste tank EP38, the basin water outlet is valved to the waste header. Using one of the basin/canal pumps, the water is pumped to waste tank EP38 while the tank level is monitored in the control room. Typically, water from only one basin is transferred at a time. Fresh deionized water from supply tank EP29 or recycle tank EP49 is available to refill the basin as it is being pumped out.

RBOF FILTER DEIONIZER WASTE

The filter deionizer loop removes particles from the basin water by use of a Porostone filter. The system also reduces the conductivity of the water by passing the water through an ion exchange tank (deionizer). The processes of draining the filter dome, backflushing the filter, and cleaning the filter with oxalic acid and caustic produce waste solutions that are sent to waste tank EP38. In these processes, the deionizer and filter are pressurized using water and air, and the waste solutions are allowed to flow to waste tank EP38 through a three-way valve connected to the waste header. To align the system to the waste tank, a normally installed blank flange must be removed to allow flow in the waste header.

WASTE FROM CASK WASH PIT

Casks received in the RBOF are usually flushed with deionized water before being placed in the basin system. This water collects in the cask wash pit. Following completion of the flushing activities, the wash pit water is transferred to waste tank EP38. Coolants from some casks are flushed directly to waste tank EP38 provided the coolant meets the limitations regarding flammability. This pit is also used for any decontamination of the cask and the removal of any road debris from transportation.

RESIN REGENERATION WASTE

The RRF receives depleted anion and cation resins from the RBOF and 100-Area deionizers. For the 100-Area deionizers, regeneration may occur in the RRF processing tanks or in the deionizer itself. These resins are separated and regenerated by adding caustic (anion resin) and nitric acid (cation resin). The mixed resin is separated based on density by a water upflow. As the anion resin flows to anion tank EP59, the waste stream drains to waste tank EP38. Regeneration by addition of acid and caustic solutions to tanks EP57 and EP59 results in waste streams that flow to waste tank EP38. The process lines are flushed after regeneration and drain to waste tank EP38. The level of waste tank EP38 is monitored in the Building 244-H Control Room during the resin regeneration/waste generation process, and communication between the Building 245-H Control Room and the Building 244-H Control Room is maintained. Regeneration occurs in discrete steps; the volume of waste produced can be estimated based on the amount of nitric acid and caustic or deionized water added to the process (Ref. 80). Typical resins are regenerated

with 5% NaOH and 5% HNO₃ at a total of 31.25 gpm for 120 minutes. This process creates about 3,750 gallons of waste. With the addition of sodium nitrite and caustic treatment (approximately 200 gallons), the total volume corresponds to 4,000 gallons.

2.5.6.2 Drainage System

Several areas in the RBOF and RRF are provided with drainage systems that flow to waste tank EP38, including the following (Figure 2.5-9):

- Hose racks
- Chemical pads No. 1 and No. 2
- Building 245-H Control Room
- Unloading platform
- Waste and resin regeneration cell sumps

HOSE BOX NO. 1 DRAINAGE

Hose box No. 1, on the west side of the RRF, provides the means for regenerating resin in the portable deionizers. The hose box No. 1 slab is sloped to a 4-inch drain. The stainless steel drain is enclosed by a 2-inch-high curb. The drain is tied to a 3-inch-diameter stainless steel header encased in concrete, which slopes to waste tank EP38. Any spills at the hose box are collected in the diked area and are drained to the waste tank (Ref. 81).

HOSE BOX NO. 2 DRAINAGE

Hose box No. 2, on the north side of the RRF, provides the means for resin disposal operations. The hose box No. 2 slab is sloped to a 4-inch drain. The stainless steel drain is enclosed by a 2-inch-high curb. The drain is tied to a 3-inch-diameter stainless steel header encased in concrete, which slopes to waste tank EP38. Any spills at the hose box are collected in the diked area and are drained to the waste tank (Ref. 81). In addition, spills on Chemical Pad No. 2 are pumped to the hose box No. 2 drain.

CHEMICAL PAD NO. 1 DRAINAGE

Chemical pad No. 1 encloses tanks EP50 and EP52, the acid storage and caustic storage tanks. The concrete pad is sloped to a 4-inch stainless steel drain similar to the hose box No. 1 drain. This drain line, along with the hose box No. 1 drain, ties into a 3-inch stainless steel header to the waste tank (Ref. 82).

CHEMICAL PAD NO. 2 DRAINAGE

This concrete pad is located northwest and adjacent to hose box No. 2. Two tanks are mounted on the pad: target wash tank EP20601 and sodium nitrite tank EP373.65. In addition, a 1,000-gallon dumpster (503-100-1) is used to replenish the inventory in the sodium nitrite tank (EP373.65). The pad is sloped to a center-mounted sump drain. This sump is connected through an isolation valve to an electric sump pump (LWS-P-001), which must be started manually (Ref. 83). The pump discharge is routed to the hose box No. 2 drain, which gravity-drains to waste tank EP38. The drain is fabricated of Type 304 stainless steel. This drain ties into the header from chemical pad No. 1 and hose box No. 2 leading to the waste tank. Chemical pad No. 2 is shown in Figure 2.5-9 (Ref. 82).

2.5.6.3 Processing and Transferring Waste (Ref. 79)

Waste that is received in waste tank EP38 is eventually transferred either to the GP evaporator through feed tank 717 or to Waste Management (WM). The destination of the waste is determined based on pH and/or radioactivity. Depending on the requirements of the receiving facility, pH ranges different from those discussed below can be transferred either to the GP evaporator (feed tank 717) or to WM. When using the low-level waste tank trailer for transferring waste, the onsite shipping requirements of Procedure 244-H-1151 (Ref. 26) are considered when determining amounts and destination of the waste.

When waste from waste tank EP38 is ready to be transferred, a sample is taken and the pH is analyzed. If the pH meets the WAC for the GP evaporator, the waste can be sent to the GP evaporator. However, if the activity of the waste does not meet the WAC for the evaporator, the waste is normally not sent to the GP evaporator but to WM.

To send waste to the GP evaporator, the level in waste tank EP38 is recorded and the tank agitator is started. Personnel in Building 211-H are contacted and requested to open the GP evaporator (feed tank 717) inlet. Next, the waste pump in waste tank EP38 is started and the transfer begins. Waste is transferred out of waste tank EP38 until about 5% of the tank volume remains.

If the waste does not meet the WAC, the waste is sent to WM after adjusting the pH. For all waste but resin regeneration waste, a sample is obtained and the pH is recorded. If needed, caustic is added to the waste in steps, and the waste is sampled until the pH is adjusted to the desired level. Sodium nitrite is then added to the waste, and the lines are flushed with deionized water. The waste is then ready to be transferred to WM (tank 21 or 23). To perform the transfer, WM is contacted and requested to open the valves on the receipt tank (tank 21-H or 23-H). The outlet valves on waste tank EP38 are opened, and the transfer pump is started. Waste is transferred from waste tank EP38 until less than 5% of the volume of the tank remains.

Before transferring resin regeneration waste, the temperature of the receipt tank must be obtained from WM. Resin regeneration waste cannot be transferred if the receipt tank temperature is

above the requirements of the WAC. The amount of sodium nitrite to be added to the resin regeneration waste is based on the receipt tank temperature. After addition of sodium nitrite, the waste can be transferred as described above. Figure 2.5-8 shows the sampling system for waste tank EP38, as well as the RBOF transfer piping for the liquid waste system.

2.5.6.4 Solid Waste

Process solid wastes from the RBOF and RRF include spent resin, bundling waste, and contaminated equipment.

SPENT RESIN

Spent resin is collected in cation tank EP57. In this tank, the resin is depleted by adding caustic. The caustic solution is circulated for approximately 2 hours for 100-Area resin and for about 4 hours for RBOF resin (Ref. 75). The resin is rinsed with deionized water and can be transferred to storage tank EP63 in the RRF. The rinse water drains to waste tank EP38.

Spent resin is transferred from tank EP57 or EP63 to a high-integrity container positioned at hose box No. 2 and generates approximately 1,650 gallons of liquid waste in waste tank EP38. The transfer is performed using plant air and the deionized water systems. Following completion of the resin transfer, the high-integrity containers are staged until disposal (Ref. 28).

BUNDLING WASTE

Cropped fuel pieces, contaminated scrap, and other bundling wastes are collected in a scrap bucket. The waste is monitored and, if acceptable, put in B-25 waste containers to be transported to the burial ground.

CONTAMINATED EQUIPMENT

When possible, equipment is decontaminated and reused (Ref. 28). For equipment for which decontamination is impractical or impossible, the equipment is packaged, monitored, and disposed of in a B-25 dumpster for burial.

2.5.6.5 Equipment and Instrumentation

WASTE TANK (EP38) (REF. 60, 74)

The waste tank located in the waste cell is a collection, neutralization, and pump tank for liquid waste discharged to the tank farm or the GP evaporator (feed tank 717) from Buildings 244-H and 245-H. The tank is 10 feet 6 inches in outside diameter by 10 feet 6 inches high with

0.25-inch wall thickness. The flat sloping bottom and flat top are 5/16 inch thick. Material is Type 304 commercial grade stainless steel. An agitator is center-bolted on a carbon steel support on the tank top. Four stainless steel baffles, measuring 10.5 inches wide by 8 feet long, on the tank wall aid in mixing. A 1-inch dip tube extends down to 2 inches above the vessel floor for measuring liquid levels. A 2.5-inch dip tube, extending to within 2 feet of the tank bottom, is the suction leg of a water eductor on the tank top. Tank capacity is approximately 6,400 gallons.

WASTE AGITATOR

The agitator is a mixer assembly with an output speed of 56 rpm. The agitator blades, shaft, and coupling are Type 304 stainless steel. The lower agitator shaft reduces from 3.5 to 2.75 inches in diameter. A four-blade agitator, 47 inches in diameter with blades 7-7/8 inches wide, is bolted to the hub. The unit is driven by a 10-hp, 440-volt, 3-phase motor.

WASTE TANK PUMP EP38.2 (REF. 84)

The waste tank contains a stainless steel submerged centrifugal pump. The liquid pumped from the tank contains some micron-sized solid particles and, occasionally, slurries of filter aid or filter precoat in less than 1% concentrations. The pump is a direct connected, vertical centrifugal pump with a flooded suction. All internal parts are commercial grade stainless steel. The pump mounting plates are carbon steel with a Type 304 stainless steel liner.

The pump columns have two bearings with an external connection for flushing and lubrication with deionized water. The motor is a 15-hp, 440-volt, 60-Hz, 1,750-rpm, flange-mounted vertical type.

2.6 CONFINEMENT SYSTEMS

This section describes SSCs or combinations of SSCs that perform a confinement function. These SSCs include facility walls and floors, process vessels and piping, and ventilation systems. Detailed structural information can be found in Section 2.4 and is not repeated here.

The RBOF handles irradiated nuclear fuel elements that are normally confined in cladding and certified concrete or steel casks. The facility also handles various waste streams with confinement provided by steel tanks. Two filtered ventilation systems in the facility provide confinement for airborne radioactivity.

The RRF receives and handles depleted exchange resins in stainless steel vessels mounted in concrete cells. Confinement for chemicals used in the RBOF and RRF is provided by steel tanks.

2.6.1 PRIMARY CONFINEMENT SYSTEMS

2.6.1.1 Fuel Cladding and Cask

Primary confinement for the RBOF is provided by intact fuel cladding. Fuels processed in the RBOF utilize a variety of cladding materials including aluminum, stainless steel, and zirconium alloy. Cladding that is metallurgically bonded to fuel, as well as unbonded cladding, is handled (Ref. 7). Fuel with suspected or known failed cladding is placed in special containers (fuel cans or test tubes - Ref. 147). Test tubes are configured to allow sampling and flushing of the fuel if required.

TEST TUBES (REF. 60)

Test tubes provide primary confinement for failed fuel elements. Test tubes are stored in a special rack adjacent to bucket storage in the #1 storage basin. There are two types of test tubes. The older type consists of a tube (14 inches in outside diameter by 13 feet 4 inches long) flanged to separate near the middle. Pipe connections to the top and bottom permit circulation and sampling of water. Tubes of this type are constructed of both aluminum and stainless steel.

The newer test tubes made of aluminum are 6 inches in outside diameter by either 5 feet 1 inch or 13 feet 2 inches long, sealed by an expandable plug in the top opening, and provided with pipe connections. The test-tube rack is constructed of aluminum I-beams and is similar to other racks. Test-tube racks are described in Section 2.4.

SPECIAL CONTAINERS

Other containers are also used in the facility to provide primary confinement for failed fuel elements. Depending on the type of fuel, failure mechanism, desired storage location, etc., Engineering may specify criteria for a storage can or may accept the fuel for canning at the RBOF. Specific ballast, sample connections, or other special requirements may or may not be desired to support safe storage in the basin. Test tubes are stored in a special rack adjacent to bucket storage in the #1 storage basin. Containers of this type are constructed of aluminum.

SHIPPING CASKS

Fuels received in the RBOF are contained in authorized shipping casks. Detailed descriptions and analysis of the casks can be found in the Safety Analysis Report for Packaging, which is required for all casks and is provided to the RBOF for each fuel shipment.

For onsite casks, several documents exist that provide details relative to the design and fabrication (Ref. 85), shipping of irradiated fuel (Ref. 86), and accident analysis of the 70-ton cask (Ref. 87). WSRC Procedure Manual 19Q (Ref. 51) specifies the requirements that the 70-ton cask must meet for shipment of radioactive materials.

2.6.1.2 Waste Storage Tank and Piping

Process waste from Buildings 244-H and 245-H flow to waste tank EP38 located in the waste cell of Building 244-H. The waste headers to waste tank EP38 consist of 3-, 4-, and 0.75-inch Type 304 commercial grade stainless steel pipes. The majority of the system piping is embedded in concrete (Ref. 88).

WASTE TANK (EP38) (REF. 74)

The waste tank is 10 feet 6 inches in outside diameter by 10 feet 6 inches high with 1/4-inch wall thickness. The flat sloping bottom and flat top are 5/16 inch thick. Material is Type 304 commercial grade stainless steel (Ref. 28). The tank has a capacity of approximately 6,400 gallons. Leaks from waste tank EP38 will be confined in the waste cell and detected by the cell dip tube and sump.

WASTE TRANSFER LINE

The waste transfer line from waste tank EP38 to the tank farm is double-walled and encased in Transite® (Ref. 7).

2.6.1.3 Filter Deionizer System

The Porostone filter (EP14) and deionizer (EP15) are located in the decontamination cell of Building 244-H. These vessels decontaminate and deionize the basin water.

RECYCLE LOOP FILTER (EP14) (REF. 7)

The filter removes 1 micron and larger solids at flow rates of 75 to 125 gpm. The filter body and tube sheet are Type 304 stainless steel. Design pressure is 125 psig. Twenty-seven Porostone filter tubes, 36 inches long, are arranged in the tube sheet to allow 1.5 inches between tube surfaces and provide a filter area of 91 ft². The tube sheet and tubes are removable as a unit. The height of the vessel is 78-5/8 inches. It has a 2-inch vent-air blow nozzle, a 3-inch outlet nozzle, a 3-inch inlet nozzle (all side-entering), and a 3-inch bottom outlet nozzle. All nozzles terminate facing downward and in the same plane to limit the amount of fixed exposed piping within the decontamination cell. The vessel bottom slopes downward. A confinement breach in the filter will be confined by the decontamination cell, which drains to the waste cell sump and activates an alarm in the Building 244-H Control Room (Ref. 28).

DEIONIZER (EP15) (REF. 22, 73)

The deionizer vessel removes radioactive and nonradioactive ions from the water flowing through it. All wetted parts are Type 304 commercial grade stainless steel. The vessel provides primary confinement for approximately 60 ft³ of mixed resins. It is 5 feet in diameter by 5 feet high with a dished top and bottom. Design pressure is 150 psi. An 18-inch manhole on top has a carbon steel closure flange. Five nozzles all terminate pointing downward in the same plane to limit the amount of fixed exposed piping in the decontamination cell. These nozzles are as follows:

- A 2-inch vent nozzle
- A 3-inch water inlet nozzle
- A 2-inch resin inlet nozzle
- A 2-inch resin discharge nozzle (all top- or side-entering)
- A 3-inch water outlet nozzle on the vessel bottom

A breach of the deionizer will be contained in the cell. Secondary confinement is also provided by the deionizer afterfilter located in the control room basement. The afterfilter collects resin particles that break through the deionizer.

RRF PROCESS VESSELS (REF. 28)

Three tanks located in the resin regeneration cells provide the primary confinement barrier for radioactive materials inside the facility. The vessels include tanks EP57, EP59, and EP63, which are the cation tank, anion tank, and resin storage tank, respectively. Each of these cylindrical vessels is fabricated of 3/16-inch stainless steel with dished tops and bottoms. A confinement breach of any of these tanks will be confined to the resin regeneration cell and will drain to the waste tank. The capacities and process functions of the tanks are discussed in Section 2.5.4.3.

2.6.1.4 Portable Deionizers and Filter

The portable 100-Area deionizer and filter mounted on the trailer on the west side of the RRF provide the primary confinement barrier for depleted resins in the deionizer and contaminants trapped in the filter.

The deionizer vessel is contained in a shielding cask (Figure 2.6-1). The deionizer vessel is fabricated from stainless steel. All piping connections to the deionizer are also stainless steel. The vessel has been tested for a hydrostatic pressure of 180 psig (Ref. 89, 90, 91). The resin from the portable deionizer is regenerated in the vessel by connecting the deionizer to line terminations from the process vessels inside the facility using flexible hoses. The resin inlet/outlet connections use a 2-inch coupling, joined to a 2-inch butterfly valve. The inlet/outlet connections are provided with a drip pan to contain leaks. Portable deionizers 1, 4, 5, 6, and 7 are not currently being used. Deionizers 2, 3, 8, and 9 are being used as resin transfer vehicles to transport monobed resin from the 100 Area to the RBOF for regeneration.

2.6.1.5 Chemical Confinement Systems

Most of the chemicals used in the RBOF and RRF are stored in stainless steel tanks on pads outside the facility. Chemicals stored outside the facility include nitric acid, caustic, sodium nitrite, target wash solution, and phosphoric acid.

The precoat and admix tanks are located in the Building 244-H Control Room.

ACID STORAGE TANK (EP50)

This 1,000-gallon portable tank is located on chemical pad No. 1 west of the facility. The tank is fabricated of Type 304 stainless steel. The tank is mounted on a 1-inch-thick concrete pad. The pad is enclosed by a 6-inch-high concrete berm for containment of spills. Leaks from the outlet valve will be contained in a drip pan that drains to the floor drain in the center of the pad (Ref. 92).

CAUSTIC STORAGE TANK (EP52)

The caustic storage tank is also a 1,000-gallon portable stainless steel vessel. The tank is mounted on chemical pad No. 1 next to the acid storage tank. Leaks from the outlet valve on this tank will drain into a 2-foot square drip pan that drains to the floor drain on the pad. The drain ties into a 3-inch schedule 40 stainless steel pipe embedded in concrete, which flows to the waste tank (Ref. 92).

ACID AND CAUSTIC PIPING

The 1-inch acid and caustic lines are encased in concrete and flow from the tanks to the southwest corner of Building 245-H where they enter the facility. The lines flow into the basement under the Building 244-H Control Room (Ref. 93).

SODIUM NITRITE TANK (EP373.65)

This tank, located on chemical pad No. 2, can contain a maximum of 1,600 gallons of 30% sodium nitrite solution. The sodium nitrite tank (EP373.65) is shown in Figure 2.5-6. It is fabricated of Type 304 stainless steel. This tank is mounted on a 1-inch-thick concrete pad equipped with a floor drain going to waste tank EP38. The pad is enclosed by a 4-inch-high concrete berm designed to contain spills from the tank. The sodium nitrite is pumped to the facility through a 1-inch stainless steel line encased similarly to the acid/caustic lines. Leaks and spills from the sodium nitrite tank are contained within the berm. A pumping station is provided to allow operations to pump down the bermed area into the waste tank (Ref. 93). The sodium nitrite tank (EP373.65) is shown on chemical pad No. 2 in Figure 2.5-9.

TARGET WASH SOLUTION TANK (EP20601)

This 1,000-gallon tank is fabricated of Type 304 stainless steel. Although this tank is not used for target cleaning and is abandoned in place, it originally received water, sodium dichromate, and phosphoric acid to prepare the solution. This tank is mounted on chemical pad No. 2 adjacent to the sodium nitrite tank.

ADMIX AND PRECOAT TANKS (EP23 AND EP17)

These two tanks are located in the Building 244-H Control Room. The admix tank (EP23) contains 3% to 5% slurries of filter aid. The tank has a capacity of 160 gallons and is fabricated of Type 304 commercial grade stainless steel. It is equipped with a split-hinged cover, which allows half of the cover to be open.

The precoat tank (EP17) is constructed similarly to the admix tank and has the same capacity (Ref. 28).

2.6.2 SECONDARY CONFINEMENT SYSTEMS

2.6.2.1 Building 244-H Structure

The building structural shell provides a secondary confinement barrier between the facility and the environment for airborne releases from the basin and car port areas. The facility walls serve to confine airborne material within the RBOF so that material will be collected and exhausted by the filtered ventilation system, which is also part of the secondary confinement barrier. The walls between the basin areas and offices also provide secondary confinement and shielding for personnel in the office areas including the control room, RCO office, change rooms, instrument shop, and lunch room. The facility walls and roof are described in Section 2.4.

2.6.2.2 Basin Walls and Floor

The walls and floors of the water-filled basins provide a secondary confinement barrier for contaminants in the basin water. The basin water itself provides the primary barrier for contaminants. The basin water is also the primary shielding mechanism for radiation exposure from normal operations and criticality events. The basin walls and floors are massive concrete structures and are described in detail in Section 2.4.

2.6.2.3 Resin Regeneration Cell

The resin regeneration cell provides secondary confinement for the process vessels located within the cell. The cell is located between the Building 245-H Control Room and the waste cell of Building 244-H. A 3-foot-thick concrete wall separates the control room and the cell. The cell shares a common 3-foot-thick concrete wall with the waste cell. The floor elevation of the resin regeneration cell is 3 feet below grade level. The legs of each vessel are bolted to the cell floor to prevent floating if the cell is flooded.

2.6.2.4 Hose Boxes

The hose boxes are located on the platform west of the RRF. The hose boxes contain the terminations from the process piping inside the RRF. Spills from the hose connections will be contained in the hose boxes and will drain to the waste tank. The hose boxes are fabricated of 16-gage Type 304 galvanized steel protected by a painted coating (Ref. 94).

2.6.2.5 Building Ventilation System

The building ventilation system (up to the point where the exhaust ducts exit the roof) provides secondary confinement for airborne releases. The RBOF/RRF ventilation systems consist of five filtered and conditioned supply systems and three exhaust systems. The ventilation system is a

“once-through” system with air flowing from areas with the least contamination potential to those with the most contamination potential. The facility is maintained at less than atmospheric pressures so that all air from the facility is filtered prior to exhaust. The air flow diagram for the RBOF and RRF is shown in Figure 2.5-7 and is taken from Drawing W233419 (Ref. 95).

FILTERS AND ROOF EXHAUST DUCTS

The filter banks for the facility are located on the south end of the roof. Two separate filter banks are provided, one serving the building exhaust and the other serving the vessel vent system. These systems are discussed in Section 2.5.5.2. The filters and the exhaust duct section on the roof are located outside the facility walls. The building exhaust provides secondary confinement, and the vessel vent system provides primary confinement. Confinement breaches in the filters are discharged directly to the atmosphere. The filters are equipped with roughing prefilters to increase the life of the filters (Ref. 96). The filter enclosures are Type 304 16-gage stainless steel. For maintenance, the enclosures are provided with airtight access doors. Exhaust ducts on the roof are also 16-gage galvanized steel measuring 33 inches in diameter.

FILTER INTERFACES

The static and differential pressure drop across the filter bank is monitored for excessive pressure drop indicating the need to replace the filter. This monitoring is necessary to prevent overpressurizing the filters, which could cause a confinement breach. Continued operation of the system with excessive pressure drop could also lead to stack exhaust backflowing into the facility. Pressure is measured in inches of water using Pitot tubes. The pressure is read from gages located in the control room.

AIR SUPPLY SYSTEMS (REF. 95, 97)

There are five supply fans for the RBOF and RRF (Ref. 98) (Figure 2.5-7):

- 1,200 cfm to the Building 245-H Control Room
- 3,750 cfm to the Building 244-H Control Room
- 4,050 cfm to the office and change areas (1,400 cfm recycle)
- 13,500 cfm to the basin areas
- 600 cfm to the waste and decontamination cells

The supply systems provide conditioned air to the facility. All air conditioning units are located on the building roof. Each unit contains intake louvers, a prefilter, heating coils, and cooling coils. Supply air flows from the roof-mounted air conditioning units down through the plenum. Supply air to the basin area, control room, and office areas is distributed using diffusers mounted approximately 12 feet above the floor.

EXHAUST SYSTEMS (REF. 95, 97, 98)

The RBOF and RRF use three filtered exhaust systems of which two are filtered. The main exhaust system discharges approximately 23,500 cfm via a stack, measuring 5 feet in diameter by 53 feet high, on the roof. The vessel vent system discharges approximately 500 cfm through a stack measuring 10 inches in diameter by 53 feet high. An exhaust fan is also provided to exhaust the restrooms (change rooms) and is not filtered.

In the receiving and storage areas, most air is exhausted at the floor level with the remainder exhausted above the 100-ton crane. In the inspection and disassembly areas, all air is exhausted at floor level.

An exhaust plenum passing through the facility roof carries approximately 15,150 cfm. This plenum receives exhaust air from exhaust ducts mounted at the floor level in the basin area (approximately 10,000 cfm). This plenum also receives 3,750 cfm from the basement and 1,400 cfm from an underground pit below the decontamination cell. The plenum passes through a duct in the roof. On the roof, 620 cfm collected by open pipe stubs in the decontamination and waste cells also flows into the duct. This exhaust duct sends a total of approximately 15,770 cfm to the roof-mounted filter bank. Two other exhaust ducts flow to the filter, including 2,950 cfm collected from the disassembly and inspection areas and 1,000 cfm from the office area for a total of 3,950 cfm, and 4,000 cfm from the main basin area and the car port area. All exhaust air passes through the filter bank on the south end of the roof. The filter unit includes prefilters and the main exhaust fan. All air flows referenced above can vary by 10% of the listed values.

VESSEL VENT SYSTEM

The vessel vent system discharges approximately 510 cfm through a stack mounted on the south end of the roof. This system provides the primary confinement for gaseous and airborne effluents in the exhaust air from the hose cabinets, regeneration vessels, and waste tank EP38. To protect against corrosion, the vessel vent exhaust ducts are fabricated from stainless steel. The ducts are sealed to be airtight similarly to the building exhaust ducts. Vessel vent air is filtered by an exhaust filter before being discharged.

2.6.2.6 System Interlocks and Interfaces

INTERLOCKS

The ventilation system is equipped with an interlock to de-energize the main supply and exhaust fans when the NIMs are activated. This interlock is intended to reduce the amount of contamination that becomes airborne during a nuclear accident. In addition, the main exhaust fan is interlocked with the supply fan to prevent operation of the supply fan when the exhaust fan is

off. This interlock is designed to ensure that differential pressures between zones do not cause air reversal (air flow from a radiologically controlled area to clean areas).

ELECTRICAL INTERFACES

All RBOF exhaust and supply fans are electrically driven. This equipment is not provided with backup power. The air balance in the facility is such that the main exhaust and supply fans, as well as the basin area and control room supply fans, must be operating to maintain sufficient air flow (Ref. 97). Maintaining sufficient air flow and thus confinement cannot be ensured during extended power outages. As a precaution, the facility is evacuated during any loss of electrical power (Ref. 99).

An electrically powered heater is installed in the vessel vent exhaust stream before it reaches the exhaust filter to minimize condensation and plugging of the filter.

STEAM INTERFACE

In the winter, building temperature is controlled by passing the air intake over coils heated by 20-psig steam. The steam supply does not affect the confinement functions of the ventilation system (Figure 2.6-2).

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2.7 SAFETY SUPPORT SYSTEMS

The purpose of this section is to describe systems that perform safety support functions. The descriptions include SSCs that perform safety functions, as well as the safety-class and safety-significant SSCs. The basin water level control system is discussed in Section 2.5.3.2.

2.7.1 AREA RADIATION MONITORING SYSTEM

The area radiation monitoring system includes the NIMs or the gamma alarm GA-6 area radiation monitors. These monitors provide coverage to indicate the following events: a release of fission products from a fuel rupture or inadvertent fuel cutting, criticality with fuel melting, or reduced shielding. The area radiation monitoring system provides audible and visual alarms to alert personnel of the potential for radiation exposure and the need for evacuation.

2.7.1.1 Functional Requirements

NUCLEAR INCIDENT MONITOR

The NIM system is the designated SRS criticality alarm system. It consists of devices that do the following:

- Detect criticality incidents with fission product releases
- Provide alarms to alert personnel to the occurrence of a criticality or high radiation levels and to initiate evacuation to minimize exposure and deter facility entry
- Provide a means to characterize the extent of the criticality accident

The description, location, design requirements, operation, and supporting analysis are discussed in detail in Chapter 6.

Six NIMs, installed in pairs at three locations, provide prompt evacuation alarms that alert facility personnel to the presence of radiation. The NIM system is capable of detecting excessive amounts or intensities of radiation, which may be due to a criticality, fuel rupture, or fuel exposure event, and of signaling immediate personnel evacuation.

Detectors are designed to alarm when subjected to a radiation field of at least 10 rads/sec and to continue to sound even though the radiation levels fall. Detectors continue to function when subjected to a criticality. Manual resets are provided outside the areas to be evacuated. A battery-power supply is provided for each NIM unit and is capable of supplying power to the unit for 24 hours. The NIM system is designed to detect an absorbed dose in free air of 20 rads (combined gamma and neutron) at a distance of 2 meters from the reacting material within 60 seconds. The alarm signal activates promptly when the dose rate at the detectors equals or

exceeds 20 rads/min at 2 meters from the reacting material. Instrument response and alarm latching occur as a result of transients of greater than or equal to 1 millisecond in duration.

The NIM unit produces an immediate evacuation signal within 0.5 seconds of activation. The NIM alarm signal is a unique sounding bell with a mid-frequency complex sound wave.

Chapter 6 indicates that a process-related criticality incident is only of concern for fuel with a water depth less than 7 feet, since an NPH-related criticality will occur only below 7 feet. Thus, credible criticality incidents in the RBOF basin will not result in a dose to personnel working over the basin (on the traveling bridge) exceeding 12 rads. Chapter 6 also evaluated the potential dose to personnel positioned throughout the RBOF other than on the traveling bridge and concluded that the water depth and the concrete wall and earth shielding around the basin precludes personnel doses of 12 rads or more. Since a 12-rad zone cannot be defined for the RBOF, use of a criticality alarm system in the RBOF is not needed.

The potential direct radiation dose to RBOF workers from a criticality in the basin is less than 12 rads; thus, exception can be taken to the need for a NIM system in the RBOF. Installation of a fission product detection system is recommended by DOE Order 420.1 (Ref. 100). Engineering is evaluating the option to deactivate (remove) the RBOF NIM system and to employ alternative instrumentation as a fission product detection system.

GAMMA MONITORING (GA-6)

GA-6 monitors are installed in the facility and are positioned to detect radiation fields for routine and nonroutine operations. Current locations are subject to change depending on the activities in progress and their impact on radiological conditions. The units are manufactured by Nuclear Measurements Corporation (NMC). The GA-6 monitors detect gamma radiation and provide audible and visual warnings. Each unit consists of the following:

- Scintillation phosphor
- Photomultiplier tube
- Transistor amplifier
- Indicating meter
- Remote recorder relays
- Alarm lights
- Horn
- Regulated low voltage supply

The monitors typically have a range of 0.1 to 1,000 mR/hr. The alarm settings are adjustable and provide for local alarms only. When the unit alarms, neither the audible nor visual alarm can be reset until the radiation level decreases below the setpoint (Ref. 101).

2.7.1.2 Interfaces with Other Systems

The NIM system interfaces with two systems: the electrical system and the ventilation system.

ELECTRICAL SYSTEM

Normal 120/208-volt AC electrical power for the NIM system is provided by emergency power panel EP513.60 located in the electrical control room. Each NIM unit also includes an internal Uninterruptible Power Supply (UPS) consisting of a 6-volt battery, which is float-charged by a rectifier. Normal AC power is used to maintain the battery fully charged and on line at all times. Emergency power panel EP513.60 also receives standby power from the tank farm normal power. The tank farm normal power is backed-up by a standby generator. If normal AC power is lost, the emergency power panel will be automatically transferred to tank farm normal or standby power. Because the tank farm diesel generator may require a minute or more before it is ready to accept a load, the NIM units will be powered by the UPS until the diesel generator comes on line. In the event the generator does not start, the UPS has the capacity to operate the NIM units for up to 24 hours. The alarm bells, however, will lose power and not operate unless the normal or backup power is available.

The terminations for energizing the alarm bells are mounted on the health and criticality cabinet in the Building 244-H Control Room.

The alarm bells are wired in two circuits. The circuits to bells in Building 244-H are run above ground using cables in metal conduits. A remote alarm is located in Building 221-H. The NIM trouble alarm also provides one of the inputs to the 244-H general building alarm in Building 221-H. Alignment of the alarm transfer switch in the Building 244-H Control Room is required to activate the remote general building alarm. RBOF personnel select the remote position when vacating the building and formally transferring control to Building 221-H (Ref. 102, 103).

VENTILATION SYSTEM

The NIMs are interlocked to shut down the main supply and exhaust fans under alarm conditions to minimize the release to the environment during a nuclear event. The interlock consists of two normally closed contacts, from a relay in the NIM alarm circuit, that are wired in series with the fan motor control circuits. Activation of the NIM alarm opens the contacts and de-energizes the fan motors (Ref. 103).

2.7.2 AIR MONITORING SYSTEMS

This section discusses systems and devices used to detect excessive radiation levels in the air. These include beta-gamma Continuous Air Monitors (CAMs) and Kanne monitors.

2.7.2.1 Continuous Air Monitoring Systems

BETA-GAMMA MONITORING

Air monitoring for beta-gamma contamination is provided by a portable NMC beta-gamma CAM system. The NMC CAM, mounted on an enclosed portable cart, consists of the following (Ref. 101):

- An end window GM detector
- A filter paper housing enclosed in a lead shield
- A recorder
- An air pumping system
- A photohelic air flow meter with high- and low-flow alarm adjustments
- An automatic scale switching count rate meter
- A high-activity alarm horn
- A red flashing alarm light
- A blue flashing air flow alarm light

The CAM is set to alarm at 3,000 counts per minute. The CAM has a relatively low air flow, so it must be placed close to or downwind from suspected sources of airborne activity. The unit does not respond to very low beta activity, and since the lead shield will not stop all response to high gamma energies, the CAM must be placed in relatively low levels of gamma background radiation. The unit is normally located on the wall outside the Building 244-H Control Room near the cask basins. The CAM alarms locally.

TRITIUM AIR MONITORING

The tritium air monitoring system for the RBOF and RRF uses one Kanne monitor, AM-3 (or RCO specified portable instrument), located in the Building 244-H Control Room to monitor airborne tritium contamination in the basin area during cask venting operations. AM-3 is not used for daily continuous monitoring of the basin area but is placed in service with source checks to ensure operability. The monitoring loop consists of a sample pump, which transfers an air sample through a 1-inch stainless steel line to the Kanne monitor. A rotameter with a low-flow alarm is installed upstream of the Kanne monitor.

In the Kanne monitor, airborne tritium becomes ionized and produces a current that is collected by an electrode. The current flows from the monitor electrode to a microammeter in the control room where it is measured. The ammeter output is recorded by a strip chart recorder. When the setpoint on the recorder is exceeded, the alarm is activated. The alarm consists of a flashing red light and a horn.

The AM-3 monitors a sample point near the disassembly basin and a sample point near the repackaging area.

In addition to alarming in the Building 244-H Control Room and being interlocked with the general building alarm, the system provides visual (flashing red light) and audible (horn) warnings in the basin area.

2.7.3 STACK MONITORING SYSTEM

Representative sampling and monitoring of the Building 244-H stacks for particulate radioactivity are accomplished by sampling the 10-inch vessel vent exhaust and the 5-foot-diameter, 53-foot-high stack. In the past, the stack monitoring system consisted of a normal range and high range monitoring system. Only the normal range monitoring system is described because the high range monitoring system is not in service.

2.7.3.1 Major Components of the System

The major components of the stack monitoring system are as follows (Ref. 101) (Figure 2.7-1):

- Two low-level gamma sodium iodide detectors coupled to photomultiplier tubes mounted in sample boxes sampling the 10- and 60-inch-diameter stacks
- Two vacuum pumps that provide motive force for the sample stream. Vacuum pump No. 1 (EP40.60) normally operates with vacuum pump No. 2 on standby.
- Alarm panel "Rack E" in the Building 244-H Control Room. This panel contains alarms for count rate meter trouble, low vacuum, vacuum pump failure, and high beta-gamma (stack and vessel vent). Strip chart recorders, which record beta-gamma releases, are also mounted on this rack. Other than the beta-gamma recorders and alarms, the alarms and equipment on Rack E are not in service.

2.7.3.2 System Operation

BUILDING EXHAUST

The flow diagram for the stack monitoring system is shown in Figure 2.7-1 (Ref. 104). The sampling point for the building exhaust duct is upstream of the filter bank and exhaust fan just before the duct enters the 5-foot-diameter, 53-foot-high stack. Equipment is installed to monitor the sample stream for tritium, beta-gamma, I-131, and fission products. Two sample boxes are provided for detection of beta-gamma and fission product particulates. The I-131 sampling and detection equipment is not in service.

PROCESS VESSEL VENT EXHAUST

The exhaust duct from the vessel vent system is sampled just before it enters the 10-inch-diameter vessel vent stack on the Building 244-H roof. This sampling system uses the two vacuum pumps that are used for the main stack. The vessel vent sample stream is monitored for beta-gamma and fission products similarly to the main exhaust duct.

INTERFACES WITH OTHER SYSTEMS

Electrical

Power for the air power vacuum blowers is from the facility Automatic Transfer Switch (ATS) supplied by normal and standby power from the tank farm. The blowers are not on standby power. The stack annunciators are provided with backup power from emergency power panel EP513 located in the electrical room (Ref. 105).

Compressed Air

The exhaust duct flow valves are solenoid-operated valves. The solenoid valves require compressed air to keep the flow valves open. Compressed air at 100 psi is received from the RBOF air compressor.

2.7.4 WASTE TANK MONITORING

Waste tank EP38, located in the waste cell of Building 244-H, receives all process liquid waste streams from the RBOF and RRF. During activities that generate liquid waste, this tank is monitored in the Building 244-H Control Room to avoid overflowing the tank.

The level in the tank is detected using upper and lower dip tubes. The dip tubes are connected to a pressure sensor located in the control room, which detects low or high level in the tank. When the pressure sensor detects high or low level, it closes a contact to the alarm circuit (red light on panel 480.10). Previously, the pressure sensors were interlocked to start the waste transfer pump when a high level was sensed and to stop the pump if a low level was detected. This function is no longer in service, and the transfer pump must be started and stopped manually.

Equipment on panel 480.10 includes the following:

- A gage that indicates the relative capacity of the tank (0% to 100%). This gage is monitored by an operator during all activities involving the waste tank.
- Selector switches for transferring waste to either feed tank 717 (GP evaporator) or the tank farm

- Waste tank agitator oil level gage
- Upper and lower dip tube purge indicators (ft³/hr)
- Alarm lights for waste cell sump high and very high levels
- Alarm light for loss of vessel vent and main exhaust fans
- Alarm light for low instrument air pressure

2.7.5 ALARMS AND SETPOINTS

Alarm response procedures (Series 244-H-ARP and 245-H-ARP) provide the operator with the actions relative to alarm configuration, system automatic functions, and corrective actions to be taken. In addition, possible causes of the alarm are specified to assist the operator in determining the initiator of the off-normal condition. Table 2.5-3 provides a summary of the alarm locations, sensing device, device setpoint, and window number (refer to the listed procedure for additional details).

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2.8 UTILITY DISTRIBUTION SYSTEMS

2.8.1 ELECTRICAL DISTRIBUTION

2.8.1.1 Primary Normal 115-kV Power

The SRS 115-kV grid is the normal source of electrical power for H Area, including the RBOF facility (Ref. 106). The SRS electrical power grid encompasses a 300-square-mile system, comprising three generating stations, 15 substations, and approximately 90 miles of 115-kV transmission lines, all coordinated in a reliable electrical power system that supplies power to all the SRS facilities. The onsite generating capacity of 110 MW is supplemented by purchased electrical power from the South Carolina Electric and Gas Company (SCE&G). The SRS grid draws power from two 115-kV transmission lines on separate rights-of-way. From the SCE&G Erquhart Station in Beech Island, South Carolina, northwest of the site, a 230-kV line ties into the plant grid on the east side of the site. This line is fed from the SCE&G V.C. Sumner Nuclear Generating Station near Columbia, South Carolina, and the SCE&G coal-fired plant at Canadys, South Carolina. The 230-kV SCE&G power is converted to 115 kV at an SCE&G transformer station, near P-Reactor area, and is delivered to the SRS 115-kV system at primary switching station 53. The second line tied into the SCE&G system at the 230/115-kV transformer station is a 230-kV line from the Georgia Power Plant Vogtle Nuclear Station in Burke County, Georgia, directly across the Savannah River from SRS.

The SRS 115-kV electrical power grid is regulated from the SRS load dispatcher's office in Building 751-A, through a computer-based supervisory control system. This supervisory control system enables the dispatcher to monitor and control key elements of the electrical power distribution. The possibility of power failures or prolonged outages, caused by faults on the grid, is minimized by the incorporation of the following features:

- The 115-kV substations and transmission lines are connected to form a network loop system.
- Switching facilities at substations permit isolation of defective transmission lines.
- With all equipment and circuits in service, the grid can sustain the loss of a 115-kV line without a power overload or interruption of essential power service.
- The 115-kV grid is protected from lightning by static wires and lightning arresters.

Electrical power is supplied to the RBOF facility from the 115-kV plant system through two parallel, redundant, 40-MVA, 115/13.8-kV transformers at substation 251-H (Ref. 107). Either transformer is capable of supplying the area electrical demand. The 115-kV double-ended supply to substation 251-H is connected to primary switching stations 52 and 53. The 115-kV primary supply is designed so that, in the event of failure of either 115-kV leg, the remaining source is unaffected and able to handle loads previously distributed between the two sources.

The H-Area substation 252-H busing and switching arrangement permits isolation of 115-kV lines, main transformers, and the 13.8-kV switchgear bus. Therefore, a single failure of equipment does not result in a prolonged outage, or curtailment of service, to the RBOF facility. The switching arrangement also allows for maintenance of H-Area equipment without curtailing production.

2.8.1.2 13.8-kV Power from Substation 252-H

Electrical power at 13.8 kV is delivered from substation 252-H to the H-Area secondary substations over six full-capacity feeder circuits. These feeders are conducted in underground cable conduit systems to pole-carried distribution systems and run overhead to the secondary substations. The 13.8-kV distribution system has loop feeder connections and switches. These provide the capability for alternate supply paths when a 13.8-kV source fails or when maintenance is required.

The 13.8-kV overhead feeders are provided with 15-kV fused switches and surge arresters at each secondary substation aerial drop.

2.8.1.3 RBOF Secondary Distribution (Figure 2.8-1)

Substation 252-10H receives power at 13.8 kV by overhead feeder 3B. Substation 252-10H is fed by a 13.8/480-volt, 1,000-kVA transformer. The substation has four 600-amp feeders to the RBOF facility. Two feeders are for Motor Control Center (MCC) No. 1 (EP515.60) and MCC No. 2 (EP515.60) located in the electrical equipment room of Building 244-H. Another feeder supplies the 100-ton crane, and the remaining feeder supplies an 80-ton air conditioning unit and heat trace transformer 901.1.

RBOF 480-VOLT DISTRIBUTION (REF. 108)

Electrical power to all equipment in the RBOF (with the exception of the 100-ton crane and 80-ton air conditioning unit) is distributed from MCC No. 1 (EP515.60) and MCC No. 2 (EP515.60).

Major loads on MCC No. 1 include the following:

- Basin pump motors
- Waste tank agitator
- Waste tank pump
- Acid and caustic pump motor
- Instrument air compressor
- RRF air compressor

- Deionized water pump motor

Major loads on MCC No. 2 include the following:

- Breathing air compressor
- Building supply and exhaust fans
- Central air conditioning units
- Main lighting panel
- Emergency distribution panel
- Chiller pump motors

BACKUP/STANDBY POWER SYSTEM

Backup/Standby power to RBOF equipment is provided by the normal H-Area tank farm electrical system. In the event of a loss of power for this system, a 275-kW diesel engine generator located in Building 254-3H (EP8-1000-1) is provided. This unit is located within the WM Facility. The feeder from the tank farm system is to an ATS located in the electrical equipment room. The transfer switch is normally fed by MCC No. 2. The load side of the ATS feeds emergency panel EP513.60 through a 480/120-volt transformer in the electrical equipment room.

LOADS ON BACKUP/STANDBY POWER

The following loads receive backup power from emergency panel EP513.60:

- NIM units
- Instrument racks in Buildings 244-H and 245-H
- Receptacles in the RCO offices
- Receptacles for personnel contamination monitors
- Stack monitoring equipment
- Master intercom

2.8.2 PROCESS WATER SYSTEM

Deionized water for all RBOF processes comes from the power house, located in Building 284-H, through a single 4-inch polyvinyl chloride line. The embedded line flows to deionized water head tank EP49, which is located on the roof of Building 244-H. A schematic of the deionized water system is shown in Figures 2.5-3 and 2.8-2.

2.8.2.1 Deionized Water Head Tank (EP49) (Ref. 28, 70)

Deionized water head tank EP49 has a capacity of 150 gallons and acts as a vent point in the deionized water supply from the power house. The tank serves as a siphon break to prevent suckback from the basins in the event of a line break or low pressure upstream. It is fabricated of 16-gage Type 304 stainless steel. It is a vertical cylindrical vessel, measuring 3 feet in diameter by 3 feet high, with a dished bottom. A cover bolted to the tank top has 1-, 3-, and 4-inch nozzles. Water from the power house enters through the 3-inch top nozzle at the center of the bottom. An inverted "J" on the 1-inch top nozzle vents the tank to the atmosphere. A float type level controller is mounted in the 4-inch nozzle. The vessel is supported by carbon steel legs.

2.8.2.2 Basin Water Distribution

Water to the RBOF basin system flows from either deionized water head tank EP49 as fresh water or recycled water from the filter deionizer system. The selection of recycled or fresh water is made using a three-way switch in the Building 244-H Control Room.

Fresh deionized water from deionized water head tank EP49 flows through a solenoid-operated control valve to a three-way valve for each basin.

2.8.2.3 Recycle Tank (EP48) (Ref. 74)

The recycle tank is mounted on the facility roof adjacent to deionized water head tank EP49. It acts as the head tank in the filter deionizer loop to prevent surges (Figure 2.5-2). The vessel is a horizontal cylinder, measuring 5 feet in outside diameter by 12 feet long, with dished heads fabricated from 3/16-inch-thick Type 304 commercial grade stainless steel. Design pressure is atmospheric. A 1-inch dip tube extending down to 2 inches above the tank bottom measures the liquid level. The vessel contains a submerged heater. Tank capacity is approximately 1,900 gallons.

The recycle tank receives water that has been decontaminated and purified from the filter deionizer system.

2.8.2.4 Resin Regeneration Water Distribution, Deionized Water Tank (EP29) (Ref. 22, 70)

Deionized process water for the RRF is received from deionized water tank EP29, which receives water from deionized water head tank EP49 (Figure 2.8-2).

Deionized water tank EP29 is located on the roof of Building 245-H and has a capacity of approximately 2,300 gallons. The Type 304 stainless steel tank is a horizontal cylindrical vessel, measuring 5 feet 6 inches in diameter by 12 feet long, with dished ends. A steam immersion heater is used to maintain water at a minimum temperature. The inlet line from tank EP49 to

tank EP29 is 3-inch polyvinyl chloride. Water from the RRF is not recycled but is drained to the waste tank.

2.8.3 STEAM DISTRIBUTION SYSTEM (REF. 109)

Steam used in the RBOF and RRF is received from the 282-H power house at approximately 300 psi (nominal 325 psi). A pressure-reducing station on the roof of Building 244-H reduces steam pressure to 100 psi. The 100-psi steam is further reduced to 25 psi for use by RBOF/RRF process systems (Figure 2.6-2).

Major users of steam in Building 244-H include hot water heaters, space heaters, and the desuperheating process. The sodium nitrite and caustic tanks use steam to maintain the tank contents at a minimum temperature. Deionized water tank EP29, which supplies deionized water to the RRF, is also steam-heated. A steam heater installed upstream of the exhaust filters in the vessel vent system assists in reducing condensation on the filters.

2.8.4 COMPRESSED AIR SYSTEM (REF. 110)

The instrument air/plant air system consists of two air compressor skids (located outside, northeast of Building 244-H) and an air dryer. Each compressor skid consists of the following: a single-stage, double-acting 7- by 5-inch compressor (H-244-IA-CMP-1 and H-244-IA-CMP-2) driven by a 20-hp motor; an aftercooler; and a receiver tank. Compressed air flows from the receiver tank through the prefilter to the plant air systems or through the air dryer prefilter, and air dryer afterfilter to the instrument air header. In case of low air pressure, a spring-loaded shutoff valve will close at 80 psig to isolate the plant air header from the system to conserve compressed air for instrument air use. The compressed air system is shown in Figures 2.8-3 and 2.8-4.

The air compressors are operated in a lead/lag configuration, which provides redundancy and improved reliability. One compressor, designated as the lead, will load when the pressure in the air receiver tank falls below 92 psig and will unload when the pressure increases to 100 psig. The lag compressor loads when the receiver pressures falls to 82 psig and unloads when the pressure reaches 100 psig. The system is normally operated in the automatic mode with the compressors running at constant speed. If the compressor runs at a constant speed for more than 20 minutes, the motor will stop until loading is required. During periods of low demand, the compressors can be operated in the manual mode.

2.8.4.1 Plant Air System

The plant air system supplies 100-psig air through a 1-inch header for use by equipment in the RBOF and RRF. The 100-psig air is reduced to 25 psig at a pressure-reducing station in Building 244-H east of the inspection basin (Ref. 111) for air-operated equipment. In Building

244-H, plant air is supplied for service stations near each basin and in the car port unloading basement and outside. In the RRF, plant air is provided for deionized water tank EP29 located on the roof of Building 245-H. Air for other RRF equipment is supplied from a separate system described in Section 2.8.4.3.

2.8.4.2 Instrument Air System

Instrument air is supplied from a 1-inch header to instrument panels through a 100 to 24 psig pressure-reducing station located in the Building 244-H Control Room. These panels distribute air to solenoid valves and other instruments in the RBOF basins and the filter deionizer loop (Figure 2.8-4).

2.8.4.3 Process Compressor (EP68) (Ref. 112, 113)

Compressor EP68 is located on chemical pad No. 1. This compressor supplies air specifically for the resin regeneration process. The compressor is driven by a 15-hp motor and has a maximum output rating of 160 cfm (although it normally operates at 7 cfm). This compressor is equipped with both intake and discharge filters, a pressure gage, and a relief system designed to operate in the 7 to 9 psi range. Compressor EP68 provides air through a 2-inch header to hose box No. 2, the cation tank (EP57), and the storage tank (EP63).

2.9 AUXILIARY SYSTEMS

2.9.1 BREATHING AIR SYSTEM (REF. 114, 115)

Activities such as fuel cropping, disassembly, and loading and unloading of casks generate airborne contamination that may require personnel performing these activities to don respiratory protective equipment. Other than breathing manifolds and connective piping, the system is external to Building 244-H. The compressor, receiver, aftercooler, and filter are located on the northeast corner of Building 244-H, adjacent to the exterior roof access stairway. The breathing air system provides a reliable source of compressed air to support these activities.

2.9.1.1 Components and System Operation

A schematic of the breathing air system is shown in Figure 2.9-1. The major components of the system are as follows:

- An electrically driven air compressor (EP811)
- An air receiver tank (EP813)
- An aftercooler (EP812) and filter (FLT-1)
- Manifold stations
- Interconnecting piping

A conceptual design change has been issued to upgrade the current breathing air system. This upgrade will replace the single 25-hp breathing air compressor with two 60-hp air compressors and associated equipment (Ref. 115).

AIR COMPRESSOR (EP811)

The breathing air compressor is driven by a 25-hp electric motor. Compressor output is approximately 100 cfm at 100 psi. Air from the compressor flows through a 4-inch schedule 40 stainless steel tubing line to the aftercooler.

AFTERCOOLER (EP812)

The aftercooler uses cooling water (taken from the domestic water system) to cool the hot compressed air to a suitable temperature. From the aftercooler, air flows to the air receiver tank.

AIR RECEIVER TANK (EP813)

The air receiver tank is equipped with a pressure relief valve set to relieve at 125 psi. Compressed air flows from the receipt tank to a filter to the breathing air header. The tank is equipped with a trap to drain condensate onto gravel outdoors.

BREATHING AIR MANIFOLD STATIONS

Manifolds along the breathing air header provide the connection points for making air hose connections to the breathing air system. Each manifold station consists of a filter, a 0- to 150-psi pressure gage, and four air hose connection sockets. Manifold stations are located in the following locations:

- In the southwest corner of the disassembly basin
- In the northeast corner of the inspection basin
- East of the cask wash pit
- On chemical pad No. 1 (two air hose connections)
- On the roof above the waste and resin regeneration cells
- On the south wall of the Building 245-H Control Room

2.9.2 CRANES AND HOISTS (REF. 28)

2.9.2.1 100-Ton Crane (Ref. 116, 117)

A bridge crane of 100-ton capacity (Figure 2.9-2) operates over the cask vehicle unloading area, yoke storage area, cask wash pit, and cask basin. This crane is used to handle the fuel shipping casks, water basin accessories, cask handling yokes, cask car crash shields, and a semi-remote impact wrench.

The crane bridge has a span of 34 feet and travels north-south on a 90-foot runway. The crane is a twin-hook type with two 50-ton hoist trolleys arranged for independent travel in the east-west direction on the bridge or for operation as a single unit electrically locked together. Vertical movement of the hooks on the hoist trolleys may be independent or may be synchronized electrically for parallel movement. Travel of the 50-ton hoist trolleys allows adjustment of the 50-ton hook spacing from a maximum of 23 feet 6 inches out of the water and 22 feet 9 inches in the basin water to a minimum of 8 feet. Maximum height of the palm of the hook above floor level is 30 feet, giving a clearance of 26 feet 6 inches over the cask basin railing, which is 3 feet 6 inches high. Vertical travel of these hooks is 60 feet. The crane is pendant-operated from the walkway on the west side of the water basins.

The bridge travels at a nominal speed of 15 ft/min. Upon crane de-energization, the solenoid brakes operate automatically.

The 100-ton crane south travel limit switch actuator bar is positioned to prevent the crane from traveling over the #2 storage basin and contacting the mechanical stops. If the 100-ton crane did contact the south end of the rail mechanical stops when loaded, the possibility of dropping the load into the #2 storage basin would exist. The limit switches are actuated by a crane stop bar mounted on the building structure to limit the crane travel by removing power to the crane motors. The location of the actuator bar is approximately 29 feet north of Column line 6 (Figure 2.9-3). DCF# E-DCF-H-01965 (Ref. 118) specifies that this location prevents the crane from contacting the mechanical stops even with the crane's momentum from a maximum speed.

A load bar with a hook (rated at 100-ton capacity) is provided to handle casks with trunnion spacings that are insufficient to be handled by the 50-ton hooks and that weigh more than 50 tons. This load bar is suspended between the lower blocks of the 50-ton hoists, and the elevation of the palm of the 100-ton hook is identical to the 50-ton hooks.

A 3-ton hoist is suspended from a monorail beneath the south girder of the cask handling crane. The primary purpose of this hoist is to provide a crane-supported, semi-remotely operated impact wrench over the cask handling area within the facility. A second 3-ton hoist is suspended from a monorail beneath the north end of the crane walkway. This 3-ton hoist is intended primarily to handle yokes and cask crash frames in the yoke storage area of the building. These hoists operate in the east-west direction on the monorails, and the maximum clearance from the palm of the hook of the impact wrench hoist to the floor is 26 feet 3 inches and from the palm of the hook of the utility hoist to the floor is 27 feet 4 inches. Vertical travel of both hoists is 35 feet, and both monorail hoists are pendant-operated. Brakes on the 50-ton hoist and the 3-ton hoists are fail-safe; if power fails, brakes automatically operate.

The crane is not provided with any equipment to move the bridge or trolleys in the event of a system failure. Crane maintenance work is performed from working platforms on the crane. Access to these crane platforms is via a barrel ladder at the north end of the building. Crane maintenance is normally performed over the yoke storage area, and this is where the crane is positioned when not in use.

2.9.2.2 Traveling Bridge (Ref. 119)

A traveling bridge (motorized platform) spans the 27-foot-wide water-filled basins, traveling north-south through the building over the basin. The traveling bridge is equipped with a 3-ton monorail on each side of the platform. The manual twin monorails are equipped to interlock with similar monorails over the canal system. The traveling bridge permits work over the cask and storage basins.

The twin monorails are suspended on a common set of end trucks. The arrangement makes up a rigid steel section that is driven on the traveling bridge rails without skewing. Suspended from

the end trucks and centered between the twin monorails is a 3-foot-wide platform or walkway with inward swinging end-gates. The total load-carrying capacity of the traveling bridge is 6,000 pounds on monorail hoists plus three persons on the platform.

The bridge has a span of 30 feet and travels on an 84-foot 5-inch railway in the facility. Spacing of the east-west-oriented twin monorails is 6 feet 6 inches center to center, with the monorail tread line elevation established at 9 feet above floor level. The bridge is powered by a geared motor (1/2 hp, 220 volt, 3 phase with squirrel cage) designed for crane service, driving the bridge at a rate of 30 ft/min. The bridge is controlled by either of two pushbutton stations, one located at each end of the walkway, and a level at the center of the south handrail. A solenoid brake on the bridge motor stops bridge movement.

Maintenance of the traveling bridge equipment is performed in the normal parking area where the traveling bridge is required to be placed upon completion of work. The normal parking area for the traveling bridge is over the wall between the #2 storage basin and the cask basin.

2.9.2.3 Monorail System (Ref. 119)

The building monorail system consists of an overhead track network with turntables on which manual trolleys are operated.

MONORAILS AND TURNABLES

An overhead track system interconnected by five turntables facilitates movement of fuel assemblies and other materials through the canals and within the working basins. The track system accommodates manual four-wheel trolleys. The monorail track is suspended from the building frame using suitable sway bracing where required. Design of the system over the canals differs from that over the disassembly and inspection basins and the repackaging basin as described below.

A single track leading to the turntables and the traveling bridge twin monorails is provided over the center of both canals. This monorail tread line is at the 9-foot elevation. The basin monorail system, including the bridge, tracks, turntables, hoists, and trolleys, is designed and rated for a distributed load of 1,000 pounds per foot of track, or 6,000 pounds distributed over a 2-foot section.

The five turntables can be turned when loaded. They are air-operated, from a pushbutton station, and turn 90 degrees, north-south to east-west, and return.

The turntables have mechanical stops to prevent a load from rolling off the table when it is turning and to prevent a load from rolling off a fixed rail when the turntable rail is not in position.

The system over the disassembly and inspection basins has a single track extension of the track over the canal system down the center of each basin. This track is rated at 1,000 pounds per foot of track for a distributed load, or 6,000 pounds per 2-foot section when that section is used only as a transfer track for a single load. Other monorails spaced 2 feet from the center track are installed at the 9-foot elevation along each side of the disassembly and inspection basins. These monorails are rated at 500 pounds per foot of track length. It is not possible to use the heavy-duty monorail trolleys on the lightweight monorail tracks over these basins. The center heavy-duty rail over the disassembly basin extends east through sliding doors 6 feet outside the building to provide for entry of heavy equipment.

The repackaging basin monorails have two tracks rated at 1,000 pounds per foot of track for a distributed load, or 6,000 pounds per 2-foot section. These tracks permit a load of 6,000 pounds on the bridge crane spanning the rails.

MOTORIZED BRIDGES

The repackaging basin has a motor-driven bridge crane supporting a moveable section of monorail track that can interlock with the canal monorail at the repackaging basin entrance. This 3-ton-capacity crane has a span of 7 feet 4 inches. It travels over the repackaging basin rails at a single speed of 40 ft/min and is stopped by a solenoid-operated motor brake. The bridge has pushbutton pendant control. Eleven-foot-long rails supported laterally beneath this bridge handle a 4-foot section of monorail that moves across the basin or interlocks with the canal system monorail. The monorail section moves at a single rate of 10 ft/min, is stopped by a solenoid-operated motor brake, and is operated by a pushbutton pendant control.

MANUAL BRIDGE

A 6,000-pound-capacity bridge is operated over the repackaging basin to the east of the motorized bridge. A trolley with a 6,000-pound-capacity hook traverses the bridge. The bridge and trolley are moved by separate chain operators.

MANUAL TROLLEYS

Trolleys are used as needed to manually move loads over the monorail system. The trolleys used are of the appropriate ratings to ensure the proper load distribution on the overhead track. Trolleys and chainfalls are checked by the operators before a lift to ensure that they are rated for the load to be carried.

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2.11 TABLES

Table 2.4-1 Glossary of Terms for RBOF Hardware

Term	Definition
Bundle	A container or holder in which fuel is handled, stored, or shipped.
Fuel storage rack	The framework for supporting fuel under water.
Grating	Gratings establish the floor of the tier. They are installed either on the storage rack beams or on top of lugs. The stamped identification designates the width of the trough guides or width of the row and position of the tier. For example, a piece of grating stamped "Mk 12 5-1/4 E" designates 5-1/4 inch width of the trough guide. The "Mk 12" designates a 12 inch row. The "E" indicates that the grating is for the east section of the row.
Guide angles	These components, found on adjustable hardware only, are bolted to the lugs. The width of the horizontal portion of the angle determines the width of the storage slot in the row when the hardware is installed. The stamped identification designates type of hardware, tier position, and width of slot formed. For example, a guide angle stamped "Mk ADJ 4-1/8 NE" designates adjustable hardware with a 4-1/8-inch width of slot positioned on the northeast side of a storage rack.
Guide plates	Guide plates are bolted or welded to the lugs and are provided with swing-down spacers and latching slots to position fuel along the row. The stamped identification designates row width, tier location, and position in the tier. For example, a guide plate stamped "Idaho 5-1/4 NE-1" would designate Idaho hardware with 5-1/4-inch distance between plates at a position on the northeast side of a storage rack at the first tier.
Isolation	Fuel is isolated if it is restrained and there is at least 12 inches of water, 60 cm of concrete, or 2 meters of air between it and any other fuel in any direction.
Lugs	Slots in this component fit over the beams in the storage rack. Other components are attached to the lugs by either bolts or welds. The lug design determines the vertical position of the hardware.
Rack hardware	The combination of guide plates, spacer plates, and gratings designed to provide specific lateral, longitudinal, and vertical spacing between fuel pieces. Hardware is also provided for storage of bundles or tubes containing several fuel pieces.

Table 2.4-1 Glossary of Terms for RBOF Hardware (continued)

Term	Definition
Removable rack tie	Removable metal latches used in place of row rack ties which are not latched or otherwise do not properly engage.
Removable spacers	Fits into slots cut in both guide plates.
Row rack tie	Metal latches that maintain horizontal spacing between adjacent I-beams.
Row width with hardware	The horizontal distance between guide plates within which the fuel pieces are supported in near-vertical position.
Row width without hardware	The center-to-center horizontal distance between rack I-beams, normally 9 or 12 inches.
Spacer plates	Latches that engage both the north and south guide plate and position fuel along the row in an east/west direction.
Spacing in row	The average center-to-center distance between fuel pieces in the same row, determined by the interval between spacer plates (also referred to as along-row spacing).
Swing-down spacers	Pivots on the south guide plate to engage a slot in the north guide plate.
Tier	One of several rows of fuel, one above another, with Tier 1 being the lowest.
Top angles	If used, these are bolted to the lugs on the top tier hardware to prevent fuel from being removed vertically.
Trough guides	Guide plates bolted or welded to the fuel storage racks and used to position the "V" trough at the bottom of a tier.
"V" trough	These are V-shaped troughs that fit between the trough guides and are intended to center the bottom of the fuel in the tier. The stamped identification designates the type of trough, width of trough, and position in the tier. Troughs are stamped for location similar to grating.

Table 2.4-2 RBOF NPH Analysis Results

Item	NPH Event	Median Capacity M_c	B_r	B_u
Steel Superstructure	Wind	191 mph	0.28	0.20
Steel Superstructure	Seismic	0.43g	1.10	0.20
Concrete Basins (Soil Liquefaction Not Considered)	Seismic	2.11g	0.38	0.50
Pre-Engineered Office Bldg Extension (Based on debris in basins due to collapse)	Wind	Does not govern	NA	NA
Girts and Siding (Based on debris in basins due to failure)	Wind	Does not govern	NA	NA
Interior CMU Walls (Based on debris in basins due to collapse)	Seismic	Does not govern	NA	NA
Exterior CMU Walls (Based on debris in basins due to collapse)	Wind	Does not govern	NA	NA
Exterior CMU Walls (Based on debris in basins due to collapse)	Seismic	Does not govern	NA	NA
Roof Purlins	Wind	Does not govern	NA	NA
IF Column	Wind	Does not govern	NA	NA
Building 245-H Nitric Acid Tank Rain Shelter	Wind	124 mph	0.27	0.40
Building 245-H Nitric Acid Tank	Seismic	3.2g	0.37	0.40

NA - Not Applicable

Table 2.4-3 Main Fuel Rack/Storage Hardware Capacity and Margin Summary

Row No.	Tier	Storage Hardware or Component	Analyzed Fuel Weight per Row ⁽¹⁾ W _F (lb/row)	Fuel Storage Capacity per Row W _{FC} (lb/row)	Margin to Code ⁽²⁾ MC= (W _{FC} /W _F)-1 MC
		E-W Seismic Capacity			
AA	1	Idaho 5 1/4, 3 Tier Type; 1st Tier Lug	22,100 ⁽¹⁾	26,490	0.20
BB	1	Idaho 5 1/4, 3 Tier Type; 2nd Tier Lug	20,000 ⁽¹⁾	25,350	0.20 ⁽⁴⁾
CC	1	Idaho 5 1/4, 3 Tier Type; 3rd Tier Lug	21,600	30,000	0.39
DD	all		5,400 ⁽³⁾	17,800	2.3
DD	1	Idaho 1 7/8; 1st Tier Lug	(0.6)5,400	18,850	4.8
DD	2	Idaho 1 7/8; 2nd Tier Lug	(0.6)5,400	18,850	4.8
DD	3	Idaho 5 1/4, 3 Tier Type; 3rd Tier Lug	(0.6)5,400	30,000	8.3
EE	1	Idaho 5 1/4, 2 Tier Type; 1st Tier Lug	5,400	17,800	2.3 ,
FF	1	Idaho 5 1/4, 3 Tier Type; 1st Tier Lug	5,400	17,800	2.3
GG	all		11,500 ⁽¹⁾	17,800	0.54
GG	1	Adj. 4 1/2 (B1.1)	5,900 ⁽¹⁾	20,700	2.5
GG	2	Adj. 4 1/8 (A2.1)	5,600 ⁽¹⁾	29,500	4.3
HH	all		9,000 ⁽³⁾	17,800	0.98
HH	1	Adj. 4 1/8 (B2.2)	(0.5)9,000	5,130	0.14
HH	2E	Adj. 4 1/2 (A2.1)	(0.5)9,000	26,490	4.9
HH	2W	Adj. 4 1/8 (A2.1)	(0.5)9,000	26,490	4.9
II	all		9,000	17,800	0.98
II	1	Adj. 4 1/8 (B1.1)	5,300 ⁽¹⁾	20,700	2.9
II	2	Adj. 4 1/8 (A1.3 & C1.3)	3,200 ⁽¹⁾	5,130	0.60
JJ	all		9,000	17,800	0.98

Table 2.4-3 Main Fuel Rack/Storage Hardware Capacity and Margin Summary (continued)

Row No.	Tier	Storage Hardware or Component	Analyzed Fuel Weight per Row ⁽¹⁾ WF (lb/row)	Fuel Storage Capacity per Row WFC (lb/row)	Margin to Code ⁽²⁾ MC= (WFC/WF)- 1 MC
JJ	1	Adj. 4 1/8 (B2.2)	4,110	5,130	0.25
JJ	2	Adj. 4 1/8 (A1.3 & C1.3)	4,890 ⁽¹⁾	5,130	0.05
KK	1	MTR-12T	9,000	17,500	0.94
LL	all		9,000	17,800	0.98
LL	1	Adj. 4 1/2 (A2.1)	(0.5)9,000	26,490	4.9
LL	2	Idaho 5 7/8	(0.5)9,000	17,800	3.3
MM	1	Idaho 5 1/4, 3 Tier Type; 2nd Tier Lug	5,400	17,800	2.3
NN	1	Hallam 5 1/4	11,000 ⁽¹⁾	17,800	0.62
OO	1	Idaho 5 1/4, 3 Tier Type; 2nd Tier Lug	19,500 ⁽¹⁾	17,800	0.17 ⁽⁵⁾
PP	1	Hallam 5 1/4	11,000 ⁽¹⁾	17,800	0.62
QQ	1	Hallam 5 1/4	5,400	17,800	2.3
RR	1	Hallam 5 1/4	9,000	17,800	0.98
SS	1-4	HFIR	6,200 ⁽¹⁾	17,800	1.88
TT	1	MTR-9TW	12,000 ⁽¹⁾	16,600	0.38
UU	1	PM-3A; Seismic Block Modification	8,500 ⁽¹⁾	17,800	1.09
VV	1	MTR-9T	9,800	14,700	0.50
WW	1	PM-3AB; 1st Tier Lug	9,000	17,800	0.98
S	1	MTR-9T	7,350	14,700	1.00
R	1	MTR-9TW	7,350	16,600	1.26
Q	1	MTR-9TW	7,350	16,600	1.26
P	1	MTR-9TW	7,350	16,600	1.26
O	1	MTR-9TW	7,350	16,600	1.26
N	1	MTR-9T	7,350	14,700	1.00
M	1	MTR-9T	7,350	14,700	1.00

Table 2.4-3 Main Fuel Rack/Storage Hardware Capacity and Margin Summary (continued)

Row No.	Tier	Storage Hardware or Component	Analyzed Fuel Weight per Row ⁽¹⁾ W _F (lb/row)	Fuel Storage Capacity per Row W _{FC} (lb/row)	Margin to Code ⁽²⁾ M _C = (W _{FC} /W _F)-1 M _C
L	1	MTR-9T	7,350	14,700	1.00
K	1	MTR-9T	7,350	14,700	1.00
J	1	MTR-9T	7,350	14,700	1.00
I	1	MTR-9T	7,300	14,700	1.00
H	1	MTR-9T	7,350	14,700	1.00
G	1	MTR-9T	7,350	14,700	1.00
F	1	MTR-9TW	7,350	16,600	1.26
E	1	MTR-9T	7,350	14,700	1.00
D	1	MTR-9T	7,350	14,700	1.00
C	1	MTR-9T	7,350	14,700	1.00
B	1	HWCTR 2.3	7,350	17,800	1.42
A	1	MTR-9T	9,000	14,700	0.63
Rack		Total Fuel Weight in Racks	Σ384,000		
		N-S Seismic Capacity			
Rack		Fuel Rack; Rack Tie Support	384,000 ⁽⁶⁾	490,000 ⁽⁶⁾	0.28
Rack		Fuel Rack; Vertical I-Beam	384,000 ⁽⁶⁾	1,356,000 ⁽⁶⁾	2.53
Rack		Fuel Rack; Wall Guide Weld	384,000 ⁽⁶⁾	619,400 ⁽⁶⁾	0.61

⁽¹⁾ The fuel weight per row is given in ECS-SSG-96-0028 and EFR-RSE-96-0115 (Ref. 40, 41).

⁽²⁾ The W_{FC} is represented by the minimum of the Hardware Capacity or the Fuel Rack Capacity.

⁽³⁾ For rows with multiple tiers, some tiers are allowed a greater share of the fuel weight than other tiers. However, the maximum weight of all the tiers must remain below the maximum fuel weight for the entire row.

⁽⁴⁾ Row BB I-beam frame load is the average of adjacent rows (20,000 + 22,100)/2=21,050 lb/row.

⁽⁵⁾ Row OO I-beam frame load is the average of adjacent rows (19,500 + 11,000)/2=15,250 lb/row.

⁽⁶⁾ Total fuel weight in all storage racks.

Table 2.4-4 Remaining Fuel Rack and Storage Hardware Capacity and Margin Summary

Row No.	Analyzed Fuel Weight Per Row ⁽¹⁾ W _F (lb/row)	Fuel Storage Capacity Per Row W _{FC} (lb/row) or All Racks W _{FC} (lb/racks)	Margin to Code ⁽²⁾ M _C = (W _{FC} /W _F)-1 M _C
Bucket Storage Racks			
0	4,200	15,580	2.71
1	4,200	15,580	2.71
2	4,200	15,580	2.71
3	4,200	15,580	2.71
4	4,200	15,580	2.71
All Racks	Σ21,000 ⁽³⁾	54,800	1.61 ⁽³⁾
Bundle Storage Racks			
5	2,200	4,550	1.07
7	2,200	4,550	1.07
9	2,200	4,550	1.07
11	2,200	4,550	1.07
All Racks	Σ8,800 ⁽³⁾	9,500	0.08 ⁽³⁾
Small Test Tube Storage Racks			
13	1,000	8,000	7.00
15	1,800	8,000	3.44
17	1,000	8,000	7.00
All Racks	Σ3,800 ⁽³⁾	9,500	1.50 ⁽³⁾

Table 2.4-4 Remaining Fuel Rack and Storage Hardware Capacity and Margin Summary
(continued)

Row No.	Analyzed Fuel Weight Per Row ⁽¹⁾ WF (lb/row)	Fuel Storage Capacity Per Row WFC (lb/row) or All Racks WFC (lb/racks)	Margin to Code ⁽²⁾ $MC = (WFC/WF) - 1$ MC
Big Test Tube Storage Racks			
19	1,200	8,100	5.75
20	1,200	8,100	5.75
All Racks	$\Sigma 2,400^{(3)}$	15,580	5.49 ⁽³⁾
#2 Storage Basin Racks			
YA	4,000	7,320	0.83
YB	4,000	7,320	0.83
YC	4,000	7,320	0.83
YD	4,000	7,320	0.83
YE	4,000	7,320	0.83
YF	4,000	7,320	0.83
YG	4,000	7,320	0.83
YH	4,000	7,320	0.83
YI	4,000	7,320	0.83
YJ	4,000	7,320	0.83
YK	4,000	7,320	0.83
All Racks	$\Sigma 44,000^{(3)}$	46,600	0.06 ⁽³⁾
#2 Storage Basin EBS Racks	150 lb/location	227.5 lb/location	0.51

⁽¹⁾ The fuel weight per row is given in T-CLC-H-00292, ECS-SSG-96-0028, and EFR-RSE-96-0115 (Ref. 39, 40, 41).

⁽²⁾ The W_{FC} is represented by the minimum of the Hardware Capacity or the Fuel Rack Capacity.

⁽³⁾ Total fuel weight in all storage racks.

Table 2.5-1 Typical Fuels Received and Stored at the RBOF

Sodium Reactor Experiment (SRE)	Riso National Laboratory (DR-3)
Oak Ridge Research Reactor (ORR)	Taiwan Research Reactor (TRR)
Rhode Island Nuclear Science Center (RINC)	Dresden
Argonne Thermal Source Reactor (ATSR)	Elk River Reactor (ERR)
Heavy Water Component Test Reactor (HWCTR)	Experimental Boiling Water Reactor (EBWR)
Savannah River Plant (MK6)	Carolinas-Virginia Tube Reactor (CVTR)
Sterling Forest Oxide Fuel (SFO)	Saxton
Missouri University Research Reactor (MURR)	Babcock and Wilcox Test Reactor (B&W)
University of Virginia (UVA)	Japan Atomic Energy Research Institute (JMTR)
Georgia Tech Research Reactor (GTRR)	Heat Transfer Reactor Experiment (HTRE)
Massachusetts Institute of Technology Reactor (MIT)	Gas-Cooled Reactor Experiment (GCRE)
Ohio State University Reactor (OSU)	Argonne National Laboratory - West, Mixed Oxide (ANL-MOX)
Greek Research Reactor (GRR)	Army Gas-Cooled Reactor (ML-1)
HOR	Experimental Breeder Reactor II (EBR-II)
Studsвик Nuclear AB (R-2)	University of Michigan (UM)
Austrian Research Centre (ASTRA)	Nereide
Saphir	Sterling Forest Fuel (SFF)
Argonne National Laboratory - Janus Reactor (ANLJ)	NOTE: Other fuel types are received at the RBOF once reviewed and approved by the Fuel Receipt and Shipping Program.

Table 2.5-2 RBOF Cask Handling

Variable	Horizontal Cask ⁽¹⁾	Vertical Cask
Loaded weight	100 tons	100 tons
Length, overall	25 feet	NA
Width, overall	9 feet	9 feet in diameter
Height, loaded cask ⁽³⁾	NA	22 feet
Elevation, top of cask over roadbed during removal from vehicle ⁽³⁾	NA	28 feet
Height, cask plus fuel during underwater loading ⁽²⁾	10 feet	33 feet

- ⁽¹⁾ Casks no more than 16 feet long may have a width of 12 feet and cask plus fuel height of 15 feet during unloading.
- ⁽²⁾ Based on 13 feet of water shielding over horizontal fuel and 10 feet of water shielding over vertical fuel; more or less may be required in individual cases.
- ⁽³⁾ With standard crane handling or use of standard type of yoke

Table 2.5-3 RBOF/RRF Alarms

Typical Alarm Panel Window Location

	1	2	3	4	5	6
A						
B						
C						

Alarm	Panel	Window No.	Device	Setpoint	ARP No.
Loss of Breathing Air	245-H	NA	Pressure Switch HR-115	94 psi	245-H-ARP-BA
NIM Trouble Stor & Cask Basin	Health Monitor Racks A & B	B3	NIM 3A or 3B	< 0.85V	244-H-ARP-NIM B3
NIM Trouble Insp. or Diss. Basin	Health Monitor Racks A & B	B4	NIM 1A, 1B, 2A, or 2B	<0.85 V	244-H-ARP-NIM B3
NIM Trouble Storage and Cask Basin	RCO Alarm Panel	B3	NIM 3A or 3B	<0.85 V	244-H-ARP-RCO B3
NIM Trouble Insp. Repack or Diss. Basin	RCO Alarm Panel	B4	NIM 1A, 1B, 2A, or 2B	<0.85 V	244-H-ARP-RCO B4
Stack Sampling System Trouble	RCO Alarm Panel	C2	Instrument Rack "E"	Various	244-H-ARP-RCO C2
Out Of Service		C5			
Lo Air Flow Basin Area	RCO Alarm Panel	E1	Flowmeter	3 scfm	244-H-ARP-RCO E1
Hi Activity Basin Area	RCO Alarm Panel	E4	Picoammeter	40.0E-05 μ Ci/cc	244-H-ARP-RCO E4
EP38 Hi Level	Panel 480.21	D4	Pressure Switch HR 233	87.5%	245-H-ARP-480.21-D4
Low Pump Suction Pressure EP29.1	Panel 480.21	C2	Pressure Switch	8 psig	244-H-ARP-480.21-C2
Hi-Low Level EP29	Panel 480.21	C1	HR 060	14.24 psig	244-H-ARP-480.21-C1
Low Level EP49	Panel 480.21	D1	HR 356	9.5 in	244-H-ARP-480.21-D1

Table 2.5-3 RBOF/RRF Alarms (continued)

Alarm	Panel	Window No.	Device	Setpoint	ARP No.
EP38 Very Hi Level	Panel 480.21	C4	Pressure Switch HR 250	91.2%	245-H-ARP-480.21-C4
Waste Cell Sump Level Very Hi	Panel 480.10	A1	Pressure Switch HR 030	66.67%	244-H-ARP-480.10-A1
Waste Cell Sump Level Hi & Lo	Panel 480.10	A2	Pressure Switch HR 225 & 226	0.67% & 26.67%	244-H-ARP-480.10-A2
Waste Tank Bearing Lo Flow	Panel 480.10	C1	Flowmeter Fischer & Porter	1.0 gpm	244-H-ARP-480.10-C1
Chiller 901.1C Fault Alarm	Panel 480.10	C4	Control Circuit Relay Panel	Various	244-H-ARP-480.10-C4
Instrument Air Low Pressure Alarm	Panel 480.10	A6	Pressure Switch HR223	≤ 18 psig	244-H-ARP-480.10-A6
Vessel Vent Exhaust Fan Loss	Panel 480.10	B3	HR 060	0.75 in of water	244-H-ARP-480.10-B3
Main Exhaust Fan Loss	Panel 480.10	B4	Pressure Switch HR 201	0.75 in of H ₂ O vacuum	244-H-ARP-480.10-B4
245 General Alarm	Panel 480.10	B5	Several	Several	244-H-ARP-480.10-B5
Expansion Tank Hi/Lo Alarm	Panel 480.10	C3	Magnetek Controls Electrode Unit	Hi - 15 in Lo - 5 in	244-H-ARP-480.10-C3
EP38 High Level	Panel 480.10	C6	HR 233	87.5% level	244-H-ARP-480.10-C6
Waste Tank Very Hi Level	Panel 480.10	A3	HR 250	91.2% level	244-H-ARP-480.10-A3
Booster Pump Failure	Panel 480.11	A2	Relay	NA	244-H-ARP-480.11-A2
EP48 High Level	Panel 480.11	A4	HR 359	8.4 psi	244-H-ARP-480.11-A4
Basin Level Alarms	480.2 to 9	A2-3, A5 - 10	HR 607-630	Hi - 6 in Lo - 14 in	244-H-ARP-480.2-480.09 / A2-A10
Loss of Breathing Air in Basin Area	Breathing Air Alarm Panel	NA	HR-101	94 psi	244-H-ARP-BA
Out Of Service		C3			

Table 2.5-3 RBOF/RRF Alarms (continued)

Alarm	Panel	Window No.	Device	Setpoint	ARP No.
245-H General Alarm	HMR Panel	C4	NA	NA	244-H-ARP-HMR-C4
Hi Activity Basin Area AM3	HMR Panel	D2	Picoammeter	40.0E-05 μ Ci/cc	244-H-ARP-HMR-D2
Lo Air Flow Basin Area AM3	HMR Panel	E2	Flowmeter	3 scfm	244-H-ARP-HMR-E2
Vessel Vent Stack High Beta Gamma	Rack "E"	A1	Victoreen CRM	1E04 CPM	244-H-ARP-IRE-A1
Bldg. Stack High Beta Gamma	Rack "E"	A4	Victoreen CRM	1E04 CPM	244-H-ARP-IRE-A4
100X G CRM Trouble	Rack "E"	B1	Victoreen CRM	Instrument malfunction	244-H-ARP-IRE-B1
108X G CRM Trouble	Rack "E"	B4	Victoreen CRM	Instrument malfunction	244-H-ARP-IRE-B4
Primary Pump Stopped	Rack "E"	C1	Pumps	Equipment failure	244-H-ARP-IRE-C1
Vessel Vent Stack Sample Flow Low	Rack "E"	C2	Flowmeter	1 scfm	244-H-ARP-IRE-C2
Bldg. Stack Sample Flow Low	Rack "E"	C3	Flowmeter	1 scfm	244-H-ARP-IRE-C3
Sample Header Low Vacuum	Rack "E"	C4	Pressure Switch 110	Low Vacuum	244-H-ARP-IRE-C4

Table 2.5-4 RBOF Basin Pump Level Setpoints

Location	Elevation
Main Basin	
Pump EP7.4 suction	-8 ft 0 in
Siphon Break	-3 ft 0 in
#2 Basin	
Pump EP7.5 suction	-8 ft 0 in
Siphon Break	-3 ft 0 in
Repackaging Basin	
Pump EP7.3 suction	-8 ft 0 in
Siphon Break	-3 ft 0 in
Cask Basin	
Pump EP7.6 suction	-8 ft 0 in
Siphon Break	-3 ft 0 in
Disassembly Basin	
Pump EP7.2 suction	-9 ft 6 in
Siphon Break	-3 ft 0 in
Inspection Basin	
Pump EP7.1 suction	-8 ft 0 in
Siphon Break	-3 ft 0 in
Transfer Canals	
Pump EP7.7 and EP7.8 suction	-8 ft 0 in
Siphon Break	-3 ft 0 in

Notes:

Top of the curb around the basins is + 4 inches.

Operator floor level is 0 inches.

Basin water level is nominally -10 to -12 inches below floor.

Low-low-level setpoint is -17 inches below floor level.

Reference: Drawing W233787

Table 2.5-5 RBOF Basin Water Limits^a

	Minimum	Maximum	Units
Basin Water pH	5.5	8.5	NA
Bulk water temperature in basin	NA	40	°C
Copper in water	NA	100	ppb
Aluminum and iron in water	NA	2000	ppb
Mercury in water	NA	100	ppb
Chloride	NA	100	ppb
Water activity	NA	7E-03	μCi/ml
Water conductivity	NA	10	μmho/cm

^a Reference 62

Table 2.5-6 RBOF/RRF Source Drawings and Vendor Blueprint Files^a

Drawing	Revision	Title	Reference
W232791	Rev. 11	Column Schedule & General Notes Steel	Ref. 8
W233662	Rev. 9	Detail 138104. Disassembly Basin Hopper and Drawings	Ref. 32
W233663	Rev. 14	Detail 138104. Disassembly Basin Hopper and Drawings	Ref. 32
W233083	Rev. 30	RBOF/RRF Basement Reflected Ceiling Plans	Ref. 35
W233076	Rev. 98	RBOF & RRF Roof Plan	Ref. 36
W233206	Rev. 12	Savannah River Plant Building 245-H Resin Regeneration Facility Equipment Arrgt., Plan and Section, Process	Ref. 47
BPF 211243	12/16/74	Model "GC" Chempump	Ref. 67
BPF 211195	10/25/61	Series H Durcopump	Ref. 68
BPF 211238	10/19/61	Precision Chemical Pump Model 601	Ref. 69
BPF 211166	9/29/61	150 Gallon SST 304 Tank	Ref. 70
BPF 211220	12/22/61	Adams Type CHS-129-91 Filter	Ref. 71
BPF 211264	11/21/61	Filter "EM"	Ref. 72
BPF 211277	12/26/61	Project 981132 & 981125 - Deionizer Vessel	Ref. 73
BPF 211178	9/29/61	Seepage Basin Tank, Waste Tank, Recycle Tank	Ref. 74
BPF 211340	9/6/73	Portable Acid Storage Tank Hatch Cover Modifications	Ref. 76
BPF 211381		Caustic Storage Tank	Ref. 77
BPF 211290	1/24/61	S.S. Pump Housings	Ref. 78
S5-2-5851	Rev. 21	RRF Hose Box Piping Arrangement	Ref. 81
C-CC-H-7846	Rev. 19	Resin Regeneration Facility No. 2 Hose Box Concrete	Ref. 82
BPF 211190	12/12/61	Assembly Parts List of the 2" D and 2 1/2" DD Vertical Pump	Ref. 84
W233799	Rev. 38	RBOF Decontamination Piping Diagram	Ref. 88
S5-1-4210	Rev. 8	Disassembly Area Portable Deionizer General Arrangement and Piping Deionizers No. 5, 6, & 7	Ref. 89
S5-1-3814	Rev. 71	Disassembly Area Portable Deionizer - General Arrangement and Piping Deionizers #2 & 3 Portable Deionizer General Arrangement and Piping Deionizers	Ref. 90

Table 2.5-6 RBOF/RRF Source Drawings and Vendor Blueprint Files (continued)

Drawing	Revision	Title	Reference
W720737	Rev. 15	SRP-Bldg-105-C, Disassembly Area, Portable Deionizer, Disassembly Process	Ref. 91
W233585	Rev. 76	RRF Piping Arrangement Plans Above Elevation 2'4"	Ref. 92
S5-2-5850	Rev. 16	RRF Piping Arrangement	Ref. 93
S5-2-5852	Rev. 5	RRF No. 2 Hose Box EP2060	Ref. 94
W233419	Rev. 47	RBOF Air Flow Diagram	Ref. 95
W233393	Rev. 36	RBOF Sections - Supply & Exhaust Sheet #1, Heating and Ventilating	Ref. 96
S5-2-10832	Rev. 15	Nuclear Incident Monitor Warning System Aerial Plot Plan	Ref. 102
S5-2-3589	Rev. 40	Improved Nuclear Incident Monitoring Facilities Electrical and Instrument Diagrams	Ref. 103
W737386	Rev. 2	RBOF Stack Sampling System P & I Diagram Piping & Instrumentation	Ref. 104
W740305	Rev. 2	RBOF Stack Sampling System Annunciator Panel Schematic Electrical	Ref. 105
W734797	Rev. 16	Single Line Electrical Drawing	Ref. 107
W232941	Rev. 70	Building 244-H, 245-H & 244-1H, RBOF and Resin Regeneration Facility, One Line Diagram (U) Electrical	Ref. 108
M-M6-H-9321	Rev. 2	Steam System Schematic	Ref. 109
W233798	Rev. 17	RBOF Service Piping Diagram	Ref. 111
S5-2-12412	Rev. 1	RRF Piping Diagram	Ref. 112
S5-2-2090	Rev. 22	Failed Fuel Element Disassembly Facilities Breathing Air System Plan, Schematic and Detail	Ref. 114
BPF 211135	8/23/62	Lock Plate Crane - 100 Ton Beam	Ref. 116
BPF 211269	3/7/62	100 Ton Crane	Ref. 117
BPF 211200	3/8/62	3 Ton Crane Schematic Wiring Diagram	Ref. 119
S5-2-7273	Rev. 16	Resin Regeneration Flow Diagram	Ref. 120
W233400	Rev. 45	RBOF Decontamination Control Diagram	Ref. 121
W233801	Rev. 46	RBOF Piping Arrangement Plan	Ref. 122
W233106	Rev. 92	RBOF Basin Discharge Pumps Interlock Diagram	Ref. 123
W233411	Rev. 43	RBOF Basin Control Diagram	Ref. 124

Table 2.5-6 RBOF/RRF Source Drawings and Vendor Blueprint Files (continued)

Drawing	Revision	Title	Reference
W233726	Rev. 47	RRF Control Diagram	Ref. 125
W233793	Rev. 9	RRF Building Instr. Arrangement & Details	Ref. 126
W233094	Rev. 47	RBOF & RRF Ground Floor Plan	Ref. 127
S5-2-5855	Rev. 17	RRF Detail Sheet 2	Ref. 128
S5-2-5849	Rev. 18	RRF Equipment Arrangement	Ref. 129
S5-2-5854	Rev. 7	RRF No. 2 Hose Box Concrete Plans and Details	Ref. 130
W233087	Rev. 90	RBOF and RRF Ground Floor & Roof Plan Drainage	Ref. 131
M-M6-H-9371	Rev. 0	RRF Flow Diagram	Ref. 132
M-M6-H-9389	Rev. 1	Acid/Caustic Systems 245-H (RBOF)	Ref. 133
M-M6-H-9509	Rev. 1	Piping and Instrument Diagram Liquid Waste System 244-H (RBOF)	Ref. 134
M-M6-H-9393	Rev. 1	Liquid Waste Stream 245-H (RBOF)	Ref. 135
M-M6-H-9348	Rev. 1	Piping and Instrument Diagram Filter Deionizer System 244-H (RBOF)	Ref. 136
M-M6-H-9510	Rev. 0	Piping and Instrument Diagram Sodium Nitrite and Transfer System 245-H (RBOF)	Ref. 137
M-M6-H-9347	Rev. 2	Piping and Instrument Diagram DI Water System 244-H (RBOF)	Ref. 138
M-M6-H-9391	Rev. 0	Deionized Water System 245-H (RBOF)	Ref. 139
M-M6-H-9390	Rev. 1	Process / Plant / Instrument Air Systems 245-H (RBOF)	Ref. 140
M-M6-H-9601	Rev. 0	Piping and Instrument Diagram HVAC System 244-H and 245H (RBOF)	Ref. 141
M-M6-H-9625	Rev. 1	Piping and Instrument Diagram Instrument Air System 244-H (RBOF)	Ref. 142
M-M6-H-9626	Rev. 1	Piping and Instrument Diagram Breathing Air Systems 244-H & 245-H (RBOF)	Ref. 143
E-E2-H-8151	Rev. 0	Electrical One Line Diagram 244-H & 245-H	Ref. 144

^a Some of the Configuration Management drawings may have "future" because of the method of updating the drawings. These drawings will be updated in the next update of the SAR.

Table 2.5-7 RBOF/RRF Supply Fans

Supply Fans			Ratings			
Switch Locations (Panels)	Equipment Number (EP)	Description	cfm	hp	rpm	volts
Main Office Hallway	H-244-502	Main Office	3,645 to 4,455	5	1,740	460
480.10-D6	H-244-503	Control Room (244-H)	3,375 to 4,125	5	1,740	460
480.10-E5	H-245-501	Control Room (245-H)	1,080 to 1,320	2	1,745	460
480.10-D7	912.1	Cell Area	540 to 660	0.75	1,750	460
480.10-D3	H-244-501	Basin Area	12,150 to 14,850	15	1,750	460

Table 2.5-8 RBOF/RRF Exhaust Fans

Exhaust Fans			Ratings			
Switch Locations (Panels)	Equipment Number (EP)	Description	cfm	hp	rpm	volts
480.10-D8	20902	Vessel Vent	459 to 561	3	1,745	440
480.10-D5	908.1	Main Exhaust	21,375 to 26,125	30	1,770	480
Men's Restroom	909.1	Office Areas	990 to 1,210	0.33	1,725	115

2.12 FIGURES

Figure 2.1.1-1 Map of RBOF Area

Figure 2.3-1 has been removed and may be found in WSRC-TR-95-0302, Nuclear Criticality Safety Manual – Spent Fuel Storage Division Facilities.

Figure 2.3-1 Plan View of the RBOF and RRF

Figure 2.3-2 Process Flow Diagram of the RBOF

Figure 2.3-2 Process Flow Diagram of the RBOF (continued)

Figure 2.3-3 Fuel Row Storage-End View

Figure 2.4-1 Elevation Schematic of the RBOF

Figure 2.4-2 has been removed and may be found in WSRC-TR-95-0302, Nuclear Criticality Safety Manual – Spent Fuel Storage Division Facilities.

Figure 2.4-2 Plan View of Storage Basins

Figure 2.4-3 has been removed and may be found in WSRC-TR-95-0302, Nuclear Criticality Safety Manual – Spent Fuel Storage Division Facilities.

Figure 2.4-3 Basement Floor Plan

2.12-9

Figure 2.5-2 Filter Deionizer System (M-M6-H-9348)

Figure 2.5-3 Deionized Water System 244-H (M-M6-H-9347)

Figure 2.5-4 RRF Flow Diagram (M-M6-H-9371)

Figure 2.5-5 Acid/Caustic Systems (M-M6-H-9389)

Figure 2.5-6 Sodium Nitrite and Transfer System (M-M6-H-9510)

Figure 2.5-7 RBOF/RRF Air Flow Diagram

Figure 2.5-8 Liquid Waste System (M-M6-H-9509)

Figure 2.5-9 Liquid Waste Stream (M-M6-H-9393)

Figure 2.6-1 Portable Deionizer

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Figure 2.6-1 Portable Deionizer

Figure 2.6-2 Steam Systems 244-H and 245-H (M-M6-H-9321)

Figure 2.7-1 Stack Monitoring Flow Diagram

RBOF/RRF Safety Analysis Report

Figure 2.8-2 Deionized Water System 245-H (M-M6-H-9391)

RBOF/RRF Safety Analysis Report

Figure 2.8-3 Process/Plant/Instrument Air Systems 245-H (M-M6-H-9390)

Figure 2.8-4 Instrument Air System 244-H (M-M6-H-9325)

Figure 2.9-1 RBOF/RRF Breathing Air System (M-M6-H-9626)

Figure 2.9-2 Schematic of 100-Ton Capacity Bridge Crane

Figure 2.9-3 South Travel Limit Switch Schematic

RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 3

HAZARD AND ACCIDENT ANALYSIS

AUGUST 1997

UNCLASSIFIED DOES NOT CONTAIN UNCLASSIFIED CONTROLLED NUCLEAR INFORMATION	
ADC & Reviewing Official:	<u>J.N. Dewes</u>
J.N. DEWES, Manager, Regulatory Programs	
Date:	<u>9/23/97</u>

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

SAVANNAH RIVER SITE

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 3

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Projects, Engineering & Construction Division
Aiken, SC 29808

SAVANNAH RIVER SITE

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ACRONYMS

AEP	Annual Exceedance Probability
ALARA	As Low As Reasonably Achievable
ALOHA	Areal Locations Of Hazardous Atmospheres
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARF	Airborne Release Fraction
ARR	Airborne Release Rate
BIO	Basis for Interim Operation
BMI	Batelle Memorial Institute (fuel cask)
BWR	Boiling Water Reactor
CAFTA	Computer-Aided Fault Tree Analysis
CFR	Code of Federal Regulations
CMU	Concrete Masonry Unit
DBA	Design Basis Accident
DF	Decontamination Factor
DOE	Department of Energy
DOT	Department of Transportation
DR	Damage Ratio
EBS	Expanded Basin Storage
EG	Evaluation Guideline
EPHA	Emergency Planning Hazards Assessment
ERPG	Emergency Response Planning Guideline
ERR	Elk River Reactor
HAD	Hazards Assessment Document
HFBR	High Flux Beam Reactor
HFIR	High Flux Isotope Reactor
HVAC	Heating, Ventilation, and Air Conditioning
HWCTR	Heavy Water Components Test Reactor
IAEA	International Atomic Energy Agency

ACRONYMS (continued)

ICR	Incremental Cancer Risk
LPF	Leak Path Factor
LWT	Legal Weight Truck
MAR	Material At Risk
MCNP	Monte Carlo N-Particle Transport Code
MIT	Massachusetts Institute of Technology Reactor
MOI	Maximally Exposed Offsite Individual
MSDS	Material Safety Data Sheet
MTR	Material Test Reactor
MURR	Missouri University Research Reactor
NAC	Nuclear Assurance Corporation
NIM	Nuclear Incident Monitor
NPH	Natural Phenomena Hazard
NRC	Nuclear Regulatory Commission
NSDS	Nuclear Safety Data Sheet
OEP	Onsite Evaluation Point
OSHA	Occupational Safety and Health Administration
PC	Performance Category
PECD	Projects, Engineering, and Construction Division
PEL	Permissible Exposure Limit
PGA	Peak Ground Acceleration
PHA	Preliminary Hazards Analysis
PMT	Plant Modification Traveler
PWR	Pressurized Water Reactor
RBOF	Receiving Basin for Offsite Fuels
RC&HP	Radiological Control and Health Physics
RF	Respirable Fraction
RFA	Reference Fuel Assembly
RHF	Reactor High Flux

ACRONYMS (continued)

RQ	Reportable Quantity
RRF	Resin Regeneration Facility
RRFC	Research Reactor Fuel Counter
SAR	Safety Analysis Report
SFO	Sterling Forest Oxide
SRE	Sodium Reactor Experiment
SRS	Savannah River Site
SSCs	Structures, Systems, and Components
SWMF	Solid Waste Management Facility
TPQ	Threshold Planning Quantity
TQ	Threshold Quantity
TSR	Technical Safety Requirement
TWA	Time-Weighted Average
USQ	Unreviewed Safety Question
USQE	Unreviewed Safety Question Evaluation
WSRC	Westinghouse Savannah River Company
ZPA	Zero Period Acceleration

3.0 HAZARD AND ACCIDENT ANALYSIS

3.1 INTRODUCTION

This chapter of the Safety Analysis Report (SAR) for the Receiving Basin for Offsite Fuels (RBOF) and adjoining Resin Regeneration Facility (RRF) (Buildings 244-H and 245-H, respectively) describes the process used to systematically identify and assess hazards for the purpose of evaluating the potential process-related, external, and Natural Phenomena Hazard (NPH) events that can cause the identified hazards to develop into accidents. Hazards analysis does the following:

- Considers the complete spectrum of accidents that may occur due to facility operations
- Analyzes potential accident consequences to the public and site workers
- Estimates the likelihood of occurrence
- Identifies and assesses associated preventive and mitigative features
- Identifies candidate safety-class and safety-significant Structures, Systems, and Components (SSCs)
- Identifies a selected subset of accidents, designated as Design Basis Accidents (DBAs) and Beyond DBAs, to be formally defined in accident analyses on the basis of a risk ranking methodology

Subsequent accident analyses evaluate these DBAs for comparison with Evaluation Guidelines (EGs) to identify and assess the adequacy of safety-class SSCs.

Department of Energy (DOE) Order 5480.23 (Ref. 1) prescribes the use of a graded approach for the effort expended in safety analysis and the level of detail presented in associated documentation. The graded approach adjusts the magnitude of the preparation effort to the characteristics of the RBOF and RRF based upon the facility hazard magnitude or severity, the facility complexity, and the facility life-cycle stage. Thus, DOE Order 5480.23 provides for developing the RBOF/RRF safety analysis based upon these three factors.

To apply the graded approach, the analysis needs only to provide sufficient calculations to demonstrate that the EGs are not exceeded. Once that is shown, additional quantitative analysis is not required. The graded approach requires a more rigorous and more thoroughly documented assessment of complex, higher hazard facilities than simple, lower hazard facilities because grading is a function of both hazard potential and complexity.

As part of the safety analysis procedure set forth by DOE Order 5480.23, a hazards analysis is a requirement for the RBOF and RRF to comprehensively identify and characterize all known hazards and evaluate them in terms of actual facility operations to assess potential risk. The

hazards analysis provides a basis for hazard classification, which in turn determines the level of safety documentation required for the facility and the level of review and approval required for the safety documentation.

This chapter includes the results of hazard identification, hazard classification, hazard evaluation, and accident analysis. Items discussed include the following:

- Methodology for hazard and accident analysis
- Identification of hazardous energy sources and hazardous material sources present
- Radiological and chemical inventories
- Facility hazard classification
- Risk evaluation of identified accident scenarios based on a qualitative or semi-quantitative assessment of consequences and frequency
- Summary of design and operational preventive and mitigative features
- Identification of planned design and operational safety improvements
- Summary of defense in depth, including identification of safety-class or safety-significant SSCs and other items needing Technical Safety Requirement (TSR) coverage
- Summary of significant worker safety features, including identification of safety-significant SSCs and any relevant TSR administrative controls
- Identification of a limited set of unique and representative accidents to be analyzed further
- Accident analysis of DBAs
- Accident analysis of Beyond DBAs
- Comparison of consequences to EGs
- Alternative methodology for dose comparison

The primary purpose of the analysis is to demonstrate that the RBOF and RRF can be operated without undue risk to onsite or offsite individuals or to the environment. The information contained within this chapter supports the conclusion that the facility can be operated safely in conjunction with the safety equipment, TSRs, and administrative control programs described elsewhere in the SAR.

3.2 REQUIREMENTS

This chapter was prepared in accordance with the format and content requirements specified in DOE Order 5480.23 and DOE-STD-3009-94 (Ref. 1, 2). The hazard identification and hazard evaluation were performed in accordance with the guidance provided in DOE-STD-3009-94 and Westinghouse Savannah River Company (WSRC) Procedure Manual 11Q (Ref. 3). The hazard classification was based upon radiological and chemical evaluations performed in accordance with the guidelines provided in DOE Order 5480.23, DOE-STD-1027-92, and WSRC-MS-92-206 (Ref. 1, 4, 5).

The quantitative accident analysis was performed in accordance with the guidance provided in DOE-STD-3009-94 and Manual 11Q. The EGs for normal events were based upon the annual consequence limits contained in DOE Order 5400.5 (Ref. 6). The EGs for accident scenarios were based upon the risk acceptability comparisons provided in Manual 11Q. The design basis criteria for natural phenomena were based upon DOE Order 5480.28, DOE-STD-1020-94, and DOE-STD-1021-93 (Ref. 7, 8, 9). The criticality evaluation was based on information obtained from ANSI/ANS-8.1-1983 and Regulatory Guide 3.35 (Ref. 10, 11).

All analyses performed for the SAR are documented in technically reviewed engineering calculations that meet the requirements of WSRC Procedure Manual E7 (Ref. 12).

DOE Order 5480.28, for the purpose of NPH design and evaluation, defines the five Performance Categories (PCs) for SSCs. DOE-STD-1020-92 provides the criteria to be used in each category. DOE-STD-1021-93 provides the criteria for categorization.

Design codes and building standards and regulations are discussed in Section 2.2.

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3.3 HAZARDS ANALYSIS

This section describes the hazards analysis performed for the RBOF and RRF. The hazards analysis is the initial analytical effort and systematically presents an analysis of potential process-related, natural phenomena, and external hazards that can affect the public, site workers, and the environment due to single or multiple failures. This analysis considers the potential for both equipment failure and human error.

The hazards analysis provides a thorough, predominantly qualitative evaluation of the spectrum of risks to the public, site workers, and the environment due to accidents involving the hazards identified. It consists of three basic analytical activities: hazard identification, hazard classification, and hazard evaluation. DOE-STD-3009-94 (Ref. 2) requires that the hazards analysis comprehensively identify potential events, event initiators, and the dominant scenarios; estimate their frequencies and consequences; identify preventive and mitigative features; and present the results in a risk matrix. Gross estimates of consequences and frequencies are performed in the hazards analysis such that attention is focused on those scenarios that are of greatest concern (i.e., highest risk). The hazard evaluation methodology represents the initial phase of accident analysis.

The hazards associated with the RBOF and RRF are both radiological and chemical in nature. Because the RBOF and RRF contain radioactive and fissile materials in such quantities that a nuclear hazard potentially exists, the facility is classified as a Nonreactor Nuclear Facility in accordance with DOE Order 5480.23 (Ref. 1).

3.3.1 HAZARDS ANALYSIS METHODOLOGY

This section presents the methodology used to systematically identify, characterize, and evaluate hazards as potential accident scenarios. In addition, the methodology used in the RBOF Hazards Assessment Document (HAD) (Ref. 13) to determine the facility hazard classification is also presented.

3.3.1.1 Hazard Identification

Hazards are primarily identified by developing comprehensive lists of all known hazardous energy sources and hazardous material sources and identifying hazardous locations. Information for identifying hazards and determining their applicability to the facility is obtained, as applicable, from the following sources:

- Existing project, safety, and environmental documents
- Design drawings and reviews
- Test plans and studies

- Facility walkdowns and equipment data
- Consultations with facility, system, or process experts

Following the identification of all potential hazardous energy sources and hazardous material sources, hazardous locations are systematically identified. First, a list of facility locations is developed. This list divides the facility into sections such as clean areas, radiological buffer areas, radiation areas, and internal contamination areas. These sections may then be further subdivided into sections based on such criteria as physical separation, common utilization, and location of distinctive hazards.

This process provides general information for the identification of hazardous energy sources and hazardous material sources and their locations. To perform the hazard evaluation, quantities of specific radionuclides and hazardous chemicals were obtained.

RADIOLOGICAL INVENTORY

The radiological/fissile hazard assessment required the development of a bounding radioactive material inventory for the RBOF, which currently stores or is expected to store a large number of fuels, including domestic and foreign research reactor fuels with varying physical, chemical, and nuclear compositions. To determine a hypothetical bounding Reference Fuel Assembly (RFA), a methodology based on the maximum amount of fissile material "burned" for each fuel assembly was used to rank the fuels. The hypothetical RFA will yield the worst-case source term for all the accident scenarios used in the accident analysis and will set the storage limitation in terms of radionuclide inventory per storage location for the RBOF.

The defined approach consisted of performing a comparison of fuel assembly characteristics, such as the following:

- Initial loading
- Type of fissionable material
- Power history
- Decay time
- Present activity
- Maximum number of assemblies stored at each location

The majority of this data was provided by the facility as spent fuel inventory tables, Nuclear Safety Data Sheets (NSDSs) (Ref. 14), Appendix A documents, and other facility records. Appendix A documents provide detailed descriptions of the material to be delivered to the Savannah River Site (SRS) in accordance with the DOE contracts with research reactors. Appendix A documents also provide the specifications and requirements that the research reactors must follow are provided by DOE/research reactors to WSRC prior to fuel shipment.

Engineering calculations provide the background information used to determine the RFA for the current and expected RBOF inventories (Ref. 15 through 20).

The hypothetical RFA that bounds the activity of any type of fuel currently stored at, or planned to be shipped to, any given RBOF location was analyzed (Ref. 21). This hypothetical RFA represents the activity of a "mocked-up" fuel that was obtained by selecting the highest activity level of each isotope from the various examined fuels. A maximum burnup of 3,000 megawatt-days per storage location was used in this safety analysis (Ref. 21). The examined assemblies are as follows:

- Missouri University Research Reactor (MURR) fuel assemblies
- Reactor High Flux (RHF) core
- Mark 42 fuel assemblies
- Dresden fuel containers

These fuels were selected as candidates for the RFA due to the high exposure they experienced and the various fuel cycles they encompass (i.e., U-233, U-235, thorium, and plutonium).

On the basis of scoping calculations performed for a Mark 22 assembly, the number of curies resulting from a typical exposure was found to vary with two factors. By varying the number of days and the reactor power, the fuel assembly's resulting isotopic inventory varied by as much as 300% with an assumed 6-month decay period. For an assumed 2-year decay period, the variation in curies reduced to approximately 65%. To properly bound the current fuel inventories, WSRC assumed, with DOE concurrence, a 2-year cooling period for all fuels shipped to the RBOF (Ref. 22). On the basis of past shipments, spent fuel is not routinely shipped to the RBOF without a 2-year decay period (defined as the time from reactor shutdown to arrival of the spent fuel at the RBOF). Therefore, a 2-year decay period for the RFA determinations (provided by the computer program ORIGEN2.1) was assumed for calculations in the SAR (Ref. 22).

The foreign research reactor spent fuels, as listed in S-CLC-H-00153 (Ref. 15), represent the potential foreign fuels for shipment to the RBOF. To characterize the foreign fuels assumed for shipment to the RBOF, a methodology similar to the above analysis (performed for the domestic fuels) was also developed in S-CLC-H-00153. The input information (e.g., fuel type, reactor power, megawatts per fuel assembly, enrichment, and burnup) was obtained from the Argonne National Laboratory database (Ref. 23). On the basis of the data available, additional calculations were performed to ultimately determine the maximum number of fuel assemblies allowed in one RBOF spent fuel location that would not exceed a 3,000-megawatt-day maximum allowable burnup.

Additionally, in developing the activity levels of each examined fuel, conservative assumptions were employed so that the specific fuel activities are overestimated. This process includes the following conservative assumptions:

- A uniform, constant power for the burnup/depletion without any cooling/decay times between irradiation cycles is assumed.
- An additional 10% pad was added to the burnup of each examined fuel to further bound the calculated radionuclide activities. This assumption accounts for uncertainties in the cross-sectional data in the ORIGEN2.1 code libraries.
- Various irradiation scenarios and neutron spectra were examined and the maximum activity level of a given isotope was selected to bound the activity level of each examined fuel.

Therefore, the total RBOF fissile material content is based on the fissile material activity for the RFA multiplied by 2,800 (i.e., the assumed number of possible fuel storage locations in the RBOF). Table 3.3-1 presents the calculation of the worst fuel unit capacity. The RFA is assumed to have a total of $1.14\text{E-}02$ Ci (or >5,500 grams) of U-235. This value is based primarily on the U-235 content of an irradiated RHF core (Ref. 24).

The developed maximum total facility radionuclide inventory is presented in Table 3.3-2 and represents the values for comparison to the hazard classification threshold criteria for Hazard Category 2 and 3 facilities.

CHEMICAL INVENTORY

The hazardous chemical inventory is based upon all the bulk chemicals stored and/or used at the RBOF and RRF. Listings from the current Material Safety Data Sheet (MSDS) notebook present all the typical chemicals and chemical products used at the RBOF and RRF. The inventory of chemicals stored in facility storage tanks or vessels was based upon the maximum design capacity of the tanks and vessels. The inventory of other bulk chemicals presented on the MSDS notebook listing is likely to vary significantly with time in terms of the specific commercial products on hand and their relative amounts. The hazardous chemical inventory list is presented in Table 3.3-3.

CHEMICAL COMPATIBILITY HAZARDS

After the chemical inventory list was developed, a hazardous chemical compatibility evaluation (Ref. 25) was conducted for the RBOF and RRF to determine if any additional hazards are created as a result of mixing the hazardous chemicals at the facility. This evaluation involved identifying hazardous chemicals at the facility, determining the possible mechanisms for mixing two chemicals, and analyzing the results to determine if a hazard is created in accordance with the Handbook of Chemical Hazard Analysis Procedures (Ref. 26). This handbook provided guidance on evaluating the result of mixing two hazardous chemicals. The chemical compatibility evaluation included the assumption that the mixing of two hazardous chemicals would occur due to the following:

- Materials mixing in process-related or interconnected drain piping or piping that leads to a common tank or sump
- Natural phenomena, causing storage vessels or containers in close proximity to breach
- Inadvertent human actions (i.e., wrong valve lineups or failure to follow process procedures)
- Equipment failure (i.e., malfunction, material failure, thermal stress, or corrosion)
- Inadequate process-related procedures

This evaluation and subsequent analysis of additional accident scenarios created as a result of chemical mixing are documented in S-CLC-H-00203 (Ref. 25).

STANDARD INDUSTRIAL HAZARDS

In addition to the radiological and chemical hazard identification, this process identifies standard industrial hazards and routinely accepted hazards. However, in accordance with DOE-STD-3009-94 (Ref. 2), hazard evaluations of standard industrial hazards and routinely accepted hazards are not included in this safety analysis. Standard industrial hazards and routinely accepted hazards are identified only to the degree that they are initiators and contributors to events resulting in radiological and chemical releases. The following characteristics are used to determine that hazards are standard industrial hazards and routinely accepted hazards:

- The hazard is routinely encountered first hand by the general public in the home, home workshop, or public areas.
- Public consensus standards exist to control the hazard.
- No evidence exists that there are public or employee concerns about the hazard beyond normal prudence.
- The hazard is subject to the Occupational Safety and Health Administration (OSHA) regulations.

Protection against standard industrial hazards and routinely accepted hazards is provided by practicing basic safety in the workplace. Such hazards are formally and systematically treated by the following programmatic elements:

- WSRC Procedure Manual 8Q (Ref. 27) defines basic sitewide safety policies and minimum requirements. This manual is augmented by detailed rules and procedures developed by departments and facilities for activities within their areas of responsibility for industrial safety.
- Industrial safety involves the detection, mitigation, management, and prevention of workplace hazards to protect against accidental death, injury, property damage, or

interruption of production. The operating philosophy at SRS is that the safety and health of employees are the first and utmost priorities. Policies are implemented at the facility level through facility-level procedures.

- During the design phase of SRS facilities or modifications to existing facilities, various design reviews contribute to standard industrial hazard identification. Application of the Management Oversight Risk Tree process and information from past operating reports also contribute to the hazard identification process.
- During facility operation, several programs ensure timely identification of standard industrial hazards. These programs include the following:
 - Self-assessments and periodic safety inspections
 - Incident investigations (formal reviews and assessments of any identified unsafe situation or incident)
 - Annual safety program reviews
 - Monthly safety meetings
 - Safety suggestion programs
 - SRS Quality Assurance Program (Ref. 28)

3.3.1.2 Hazard Classification

The purpose of the hazard classification is to evaluate the potential radiological and chemical hazards associated with the facility and to determine the proper hazard classification based on the guidelines provided by DOE Order 5480.23, DOE-STD-1027-92, and WSRC-MS-92-206 (Ref. 1, 4, 5), as applicable. The focus in the hazard classification is on the impact of maximum energy or material source term release, without credit for controls or regard for event frequency. The major objective of the hazard classification is to provide the input data for a graded approach to develop the facility SAR in accordance with DOE Order 5480.23. To satisfy this objective, event consequences are calculated using conservative methods without consideration for any engineered features or administrative controls.

DOE Order 5480.23 defines three hazard categories for facility classification as follows:

- A facility is designated as Hazard Category 1 if the analysis shows the potential for significant offsite consequences.
- A facility is designated as Hazard Category 2 if the analysis shows the potential for significant onsite consequences.
- A facility is designated as Hazard Category 3 if the analysis shows the potential for significant localized consequences only. A facility that does not exceed the Hazard Category 3 threshold criteria but still possesses some amount of radioactive material is considered to be a Radiological Facility.

The analysis conducted to determine the facility hazard category is performed without credit taken for engineered features or administrative controls. Examples of engineered features or administrative controls are those specific facility features (not including site location), such as building containment, stacks, equipment, systems, actions, or operating conditions, that are established to control risk. This approach allows credit to be taken for existing location and the physics of a material release and dispersion, but not for containment, confinement, shielding, protection systems, administrative controls, or human activities.

The radiological evaluation is accomplished by comparing the inventory of each radionuclide to the threshold quantities provided in Table A.1 of DOE-STD-1027-92. The chemical evaluation begins with a comparison of each chemical to the Reportable Quantities (RQs) of Title 40 Code of Federal Regulations (CFR) Part 302.4, the Threshold Planning Quantities (TPQs) of 40 CFR 355, and the Threshold Quantities (TQs) of 29 CFR 1910.119 (Ref. 29, 30). Analysis is not required for any chemicals that do not exceed the RQs, TPQs, or TQs because chemicals in quantities less than those listed are generally accepted by the public. Following the initial screening process, the remaining chemicals are evaluated in accordance with the criteria of WSRC-MS-92-206.

The criteria for determining the hazard categorization of nuclear facilities, based on the radiological and chemical evaluations, are established as follows:

Hazard Category 1

- The facility has the potential for significant offsite consequences based on total curie content, potential material forms, and maximum energy for dispersion available. Only Category A Reactors or facilities designated by the Program Secretarial Officer are designated as Hazard Category 1 in accordance with the directions of DOE-STD-1027-92 (Ref. 4).
- A postulated release of any hazardous chemical results in an airborne concentration greater than its Emergency Response Planning Guideline 3 (ERPG-3) value at the site boundary (Ref. 5).
- For chemical compounds that have been identified as confirmed or suspected carcinogens, the calculated Incremental Cancer Risk (ICR) is greater than or equal to $2.00\text{E}+00$ at the site boundary (Ref. 5).

Hazard Category 2

- The quantity of any radionuclide exceeds the Hazard Category 2 threshold quantity provided in Table A.1 of DOE-STD-1027-92 (Ref. 4). This quantity is based on 1 rem at a distance of 100 meters (onsite) as determined in 10 CFR 30 (Ref. 31) and as modified by DOE.

- The minimum critical mass limit for any fissile material, as specified in ANSI/ANS-8.1-1983, is exceeded (Ref. 4, 10).
- The total quantity of mixed fission products, where the individual radionuclides have not been determined, is greater than 1,000 Ci (Ref. 4).
- Where there are combinations of radioactive materials, the sum of the ratios of the quantity of each radionuclide to the Hazard Category 2 threshold quantity exceeds 1 (Ref. 4).
- A postulated release of any hazardous chemical results in an airborne concentration greater than its ERPG-3 value onsite at a distance of 100 meters (Ref. 5).
- For chemical compounds that have been identified as confirmed or suspected carcinogens, the calculated ICR is greater than or equal to 2.00E+00 onsite at a distance of 100 meters (Ref. 5).

Hazard Category 3

- The quantity of any radionuclide exceeds the Hazard Category 3 threshold quantity provided in Table A.1 of DOE-STD-1027-92 (Ref. 4). This quantity is based on 10 rem at 30 meters based on a 24-hour exposure.
- Where there are combinations of radioactive materials, the sum of the ratios of the quantity of each radionuclide to the Hazard Category 3 threshold quantity exceeds 1 (Ref. 4).
- A postulated release of any hazardous chemical results in an airborne concentration greater than its ERPG-3 value at the facility (Ref. 5).
- For chemical compounds that have been identified as confirmed or suspected carcinogens, the calculated ICR is greater than or equal to 2.00E+00 at the facility (Ref. 5).

3.3.1.3 Hazard Evaluation

The hazard evaluation constitutes the primary focal point of the hazards analysis. The hazard evaluation provides the detailed information that allows the development of specific events and scenarios associated with hazardous material releases and the estimation of their frequencies and consequences. The majority of information required to perform the hazard evaluation was obtained through system-by-system reviews of facility documentation, facility walkthroughs, and consultations with facility personnel. The hazard evaluation was performed in accordance with the requirements of DOE-STD-3009-94 and WSRC Procedure Manual 11Q (Ref. 2, 3).

The primary goal of the hazard evaluation is to identify and evaluate specific events associated with hazardous material releases. The hazard evaluation is presented in tabular form and includes the following information:

- Event
- Causes
- Preventive features (design and administrative)
- Frequency level
- Method of detection
- Mitigative features (design and administrative)
- Consequences (impact on other systems and health and safety)
- Risk ranking

Each of these features is discussed in the following sections.

EVENT

The first step in the hazard evaluation process is to identify events that could result from the facility hazards. Hazardous energy sources, hazardous material sources, and their locations are evaluated to develop a complete list of potential events. The events are then included in the hazard evaluation tables. These tables include an event number and postulated event description. The event number is simply a method for identifying each event. The event categories and corresponding codes used in the hazard evaluation tables are as follows:

<u>Code</u>	<u>Event Categories</u>
E-1	Fire/Explosion
E-2	Loss of Containment
E-3	Loss of Confinement
E-4	Injury to Workers
E-5	Criticality
E-6	Interfaces
E-7	Waste Disposal
E-8	Maintenance
E-9	Extrinsic Hazards
E-10	Natural Phenomena

Each process was then located within the process areas defined in Section 3.3.2.1 and was analyzed against the event categories (E-1 through E-10), one at a time, postulating event scenarios for the applicable hazardous energy sources and hazardous material sources listed in Tables 3.3-5 and 3.3-6, respectively. The identified event scenarios cover the spectrum of possible events for a given hazard from small consequence events, for which existing procedures

or equipment is acknowledged to provide adequate protection, to reasonable worst-case conditions, in terms of both event frequency and consequence magnitude.

The postulated event description includes information concerning the type of event, location, system, interfaces with other systems, equipment, and operation characteristics that are involved in the event. This information is provided in the hazard evaluation tables to clearly define the scenario that is being evaluated.

It should be noted that the event numbers in Tables 3.3-13 through 3.3-17 may not be consecutive. Tables 3.3-13 through 3.3-17 are sorted by event class, then event number.

CAUSES

The second step in the hazard evaluation process is to determine the cause of each postulated event. The hazard evaluation tables contain a description of these causes. The majority of the causes were determined by extensive in-depth reviews of the systems, historical event data, and other sources of information (e.g., existing safety documentation). The causes of the postulated event scenario include postulated failures, operator errors, analysis errors, and environmental and operating conditions. The causes of a postulated event are clearly identified to support the frequency evaluation for the event.

PREVENTIVE FEATURES

The third step in the hazard evaluation process is to identify the preventive features associated with the facility that would reduce the frequency of a postulated release of hazardous material. The hazard evaluation tables include a description of the preventive features. Design and administrative features (e.g., procedures, policies, and programs) are included as preventive features. Any facility design and/or administrative features that are credited with reducing the frequency of a postulated event are identified in the hazard evaluation tables (Tables 3.3-13 through 3.3-17) by underlining. Preventive features implicitly credited in reducing the frequency of postulated accident conditions include the following:

- The worst-case assembly (heat load) (refer to Event 8a) that the RBOF currently stores or is expected to store has a heat load equivalent to a MURR assembly with 135 watts of decay heat. Fuel shipped in casks without coolant is not subject to this requirement due to the small heat load.
- The RBOF fissile material content is based on the fissile material activity of the RFA multiplied by 2,800 (the assumed number of possible fuel storage locations in the RBOF). The worst-case assembly is assumed to have a total of 1.14E-02 Ci (or >5,500 grams) of U-235. The value is primarily based on the U-235 content of an irradiated RHF core. Procedural controls require the preservation of these values.

- A basic fire protection program that controls the combustible loading in the RBOF and RRF is implicitly credited in minimizing the fire hazard.
- A basic worker competence level is maintained by a worker training and qualification program.

FREQUENCY LEVEL

The fourth step in the hazard evaluation process is to determine the frequency level for the occurrence of a postulated event resulting from the causes identified. The hazard evaluation tables include an estimate of the frequency level. Determining frequency required using several different methodologies for the various postulated events. When available, failure rate data, historical event data, and other sources of information (existing safety documentation) were used to determine event frequencies for process-related postulated events. For process-related events, information specific to the RBOF and RRF was obtained from site data banks, and a frequency determination was made as discussed in S-CLC-H-00203 (Ref. 25).

For process-related postulated events that have not occurred in the RBOF or RRF, a method based on the Rare Event Theory (Ref. 32) was used for determining initiator frequency. This method is based upon mathematical modeling in which the estimated number of failures occurring over time is approximated. Standard error factors, interval estimate, and confidence bound areas are also used in the mathematical model. In simple form, if an event has not occurred in t years, a conservative upper limit of the failure frequency (at the 95% confidence level) is given by the following:

$$\text{Upper high limit frequency} = 3/t \quad (\text{Eq. 3.3-1})$$

This method is acceptable for hazard evaluations because it provides a conservative estimate of frequency to provide a high-confidence estimate. Therefore, for all postulated event scenarios identified that historically have not occurred during the operational period of the RBOF and RRF, an initiator frequency of $9.38\text{E-}02/\text{yr}$ was used.

For NPH events, the frequency of occurrence is based upon guidance provided in DOE Orders 5480.23 and 5480.28, DOE-STD-1020-94, and DOE-STD-1021-93 (Ref. 1, 7, 8, 9).

DOE Order 5480.23 requires the use of a graded approach in performing safety analyses and evaluations of DOE facilities for normal operating and accident conditions, including accidents caused by NPH events. DOE Order 5480.28 uses this graded approach and requires, for the purpose of NPH design and evaluation, placing the SSCs comprising the DOE facilities into one of the five PCs. DOE-STD-1021-93 provides the criteria to be used for such categorization of SSCs and recommends systematic procedures to implement these criteria. The NPH evaluation of Building 244-H classified the reinforced-concrete basins, fuel assemblies, and fuel racks as PC-3 because of the radiological hazards (Ref. 33). The RBOF/RRF's steel framed enclosure structure, roof, walls, cranes, and crane support were classified as PC-2. The SSCs were evaluated for interaction concerns, using the loading criteria of the higher PC SSC, and ultimate

allowable stress (i.e., the building structure or cranes interacting with the fuel assemblies and fuel racks). Therefore, the Annual Exceedance Probabilities (AEPs) for PC-3 are assumed for the hazard evaluation portion of the SAR relative to NPH events.

Regardless of which method is used (i.e., historical data, Rare Event Theory method, or frequencies from DOE-STD-1020-94), as required by DOE-STD-3009-94 (Ref. 2), uncertainties in frequency values are accommodated by erring in the conservative direction from best-estimate values. Table 3.3-7 provides the frequency levels and their corresponding codes used in the hazard evaluation tables (Tables 3.3-13 through 3.3-17). In addition to the frequency codes, the hazard evaluation tables contain one of the following source codes, identifying how the frequency was determined:

<u>Code</u>	<u>Source</u>
1	Reference 25: SEPR Data Bank (200-Area Fault Tree Data Bank) Review
2	RBOF Systems Analysis (Ref. 34) and/or Safety Analysis (Ref. 35)
3	Rare Event Theory method of frequency determination (Ref. 32)
4	UCRL-15910/DOE-STD-1020-94 (Ref. 36, 8)

METHOD OF DETECTION

The fifth step in the hazard evaluation process is to identify the method of detection for a postulated event. The method of detection was determined through system-by-system reviews of facility documentation, facility walkthroughs, and discussions with facility personnel. The hazard evaluation tables provide the methods of detection (e.g., alarms, instrumentation, sensors, and facility personnel).

MITIGATIVE FEATURES

The sixth step in the hazard evaluation process is to identify the mitigative features associated with the facility that would reduce the consequences of the postulated event. Mitigative features were identified by reviewing selected facility procedures and historical safety documentation and by interviewing facility personnel. The hazard evaluation tables include a description of the mitigative features. Design features and administrative procedures are included as mitigative features. Any facility design or administrative features that are credited with reducing the consequences of a postulated event are identified in the hazard evaluation tables (Tables 3.3-13 through 3.3-17) by underlining. Preventive features implicitly credited in reducing the consequence of postulated accident conditions include the following:

- The worst-case fuel assembly (radioactive fuel material inventory) that the RBOF basin currently stores or is expected to store has at least a 2-year decay period or equivalent.

- The RBOF fissile material content is based on the fissile material activity of the worst-case fuel assembly multiplied by 2,800 (the assumed number of possible fuel storage locations in the RBOF). The worst-case assembly is assumed to have a total of $1.14\text{E-}02$ Ci (or $>5,500$ grams) of U-235. The value is primarily based on the U-235 content of an irradiated RHF core.
- The radioactive and hazardous material inventories documented in the RBOF HAD (Ref. 13) and listed in Tables 3.3-2 and 3.3-3 are not exceeded.

CONSEQUENCES

The seventh step in the hazard evaluation process is to determine the consequence level for the postulated event in terms of health and safety, as well as the estimated impact on other hazard evaluations. Considering the hazardous energy source and/or hazardous material source (Tables 3.3-3 and 3.3-4) associated with the postulated events, system-by-system reviews of the RBOF and RRF were performed as necessary to determine the impact that the event would have on other systems. These impacts are listed in the hazard evaluation tables.

The postulated events cover the spectrum for a given hazard, from small consequence events to reasonable worst-case conditions. Thus, conservative assumptions are incorporated into the evaluation, while incredible, worst-case assumptions are not included.

In accordance with DOE-STD-3009-94 (Ref. 2), the consequence levels are based on a qualitative assessment. Tables 3.3-8 and 3.3-9 provide the guideline definitions on consequence severity that are used in the hazard evaluation tables.

The 500-mrem limit at the site boundary was chosen as a high consequence event to ensure that accident scenarios (credible events) with consequences that could exceed the offsite radiological EGs are evaluated quantitatively to identify safety-class functions that are required to prevent and/or mitigate the consequences of such postulated accident scenarios.

Permissible Exposure Limit - Time-Weighted Average (PEL-TWA) at the site boundary was chosen as a high consequence event to ensure that accident scenarios (credible events) with consequences that could exceed the offsite chemical EGs are evaluated quantitatively to identify safety-class functions that are required to prevent and/or mitigate the consequences of such postulated accident scenarios.

The moderate and low offsite radiological and chemical consequence guidelines were not applied in the hazards analysis methodology as a result of the conservative values (500 mrem and PEL-TWA, respectively) employed for defining the high consequence events.

High consequence level guidelines for the onsite worker were not defined on the basis of the conservatism used in assigning high consequence levels for the offsite EGs. Events that result in life-threatening injuries due to a release of hazardous radiological or chemical material are conservatively defined as events that result in a (1) radiological dose of greater than 25 rem or (2)

an air concentration greater than the ERPG-2 value for the applicable released chemical. ERPG-2 is defined as the maximum airborne concentration below which nearly all individuals could be exposed for up to one hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair an individual's ability to take protective action.

Events that result in non-life-threatening injuries or no permanent disabilities due to a release of hazardous radiological or chemical material are conservatively defined as events that result in a (1) radiological dose of less than or equal to 25 rem but greater than 500 mrem or (2) an air concentration less than or equal to the ERPG-2 value but greater than ERPG-1 value for the applicable released chemical.

At the time of SAR development, these criteria were conservatively assumed with respect to later approved guidance documents (i.e., hazard evaluation procedures (WSRC-RP-95-1001 [Ref. 37]) and Procedure 2.25 of WSRC Procedure Manual E7 [Ref. 12]).

RISK RANKING

The final step in the hazard evaluation process is to determine the risk ranking for each postulated event. Risk rankings prioritize accident scenarios for review or further analysis and are included in the hazard evaluation tables.

Risk rankings are associated with sectors of the risk matrix (Table 3.3-10) in a manner that prioritizes accident scenarios for review or further analysis primarily in terms of the risk that they present, with the consideration that scenarios in the low consequence sectors should represent no more than a marginal concern, regardless of the frequency. Risk rankings are identified and distinguished based upon consequences to the Maximally Exposed Offsite Individual (MOI) and site workers.

The risk ranking for each postulated event is determined by locating the event on a risk matrix based upon the consequence level and frequency level of the event. The risk ranking for each event is identified based upon the consequences associated with the MOI and the onsite worker.

RESULTS

The purpose of the hazard evaluation process is to identify those events that require further analysis. Events that pose a considerable onsite and offsite impact on people and the environs (i.e., have a risk rank of 1, 2, or 4) are required to be analyzed further in Section 3.4. Events that pose a considerable threat to the onsite individual and only minor offsite impact (i.e., have a risk rank of 3, 5, or 7) require qualitative (or semi-quantitative) evaluation, primarily to address worker safety issues. Engineered and administrative preventive and mitigative features that are used to reduce the frequency and/or consequences of postulated accidents below the guidelines in Tables 3.3-8 and 3.3-9 are considered candidate safety-class or safety-significant items. Events that pose only a minor threat to the onsite individual and a negligible offsite threat (i.e., have a

risk rank of 6, 8, or 9) may require safety-significant functions, but not in most cases. Events that have negligible consequences (i.e., have a risk rank of 10, 11, or 12) require no further analysis. Finally, events that are considered beyond extremely unlikely (i.e., have a frequency less than or equal to $1.0E-06/\text{yr}$) have a risk rank of 13.

Hazardous Chemical Compatibility

In this section, scenarios as a result of two hazardous chemicals being mixed are presented. Chemical inventories at the RBOF and RRF were determined from WSRC-TR-94-054 (Ref. 13). It should be noted that Revision 1 of WSRC-TR-94-054 was used for the chemical inventory since the hazard classifications were based on this inventory. WSRC-TR-94-054 has since been revised (Revision 2) and will be referenced as appropriate. The first step of this analysis identified chemicals that could possibly be mixed in the event of an inadvertent human error, inadequate procedures, natural phenomena, or a process-related action. The possible mixing mechanisms identified include the following:

- Chemicals stored together that are mixed as a result of a container rupture. A facility walkdown determined the storage locations of the hazardous chemicals at the RBOF and RRF. These storage locations are documented in Table 3.3-3.
- Facility process and/or drain piping or interconnected piping that drains to a common vessel, resulting in chemical mixing. RBOF/RRF process diagrams depicting flow paths and drain piping were reviewed to determine if mixing could occur, either by mixing in a common pipe or vessel or by a valve being out of position.
- Mixing caused by inadvertent human error during process-related activities at the RBOF and RRF
- Equipment failure caused by malfunction, material failure, thermal stress, or corrosion that leads to mixing
- Inadequate procedures or an error in a process procedure step resulting in mixing
- Natural phenomena, causing the rupture of storage containers/tanks of chemicals in close proximity

Based upon the guidelines established in the Handbook of Chemical Hazard Analysis Procedures (Ref. 26), a variety of consequences may be expected to occur when two materials are combined. The mixing may produce the potential for any of the following:

- Heat
- Fire
- Explosion
- Violent polymerization
- Toxic gases

- Flammable gases
- Innocuous gases
- Solubilization of toxic substances

Table 3.3-3 provides the reactivity group name and reactivity group number of the chemicals found at the RBOF and RRF, based upon the methodology provided in the handbook.

A hazardous chemical evaluation matrix (Ref. 25) was developed to assess the likelihood of the mixing of the chemicals listed in Table 3.3-3. Based upon the results of this evaluation, the following sections present the likely hazardous chemical combinations that could occur at the RBOF and RRF.

Sodium Hydroxide and Cation/Anion Resin

Sodium hydroxide is used to regenerate anion resins and to deplete cation resins at the RBOF and RRF. No hazards are identified with the mixing of these two chemicals, and no further analysis is required.

Nitric Acid and Cation/Anion Resin

Nitric acid is used to regenerate cation resins and to deplete anion resins. The RBOF/RRF hazards analysis (Ref. 25) determined that, during regeneration of cation resin, an overpressurization (explosion) could occur in the resin regeneration tank if undiluted nitric acid is added to the cation resin being regenerated. Table 3.3-15 discusses this scenario as Event 400a.

Nitric Acid and Sodium Hydroxide

Nitric acid is stored in tank EP50, and sodium hydroxide is stored in tank EP52 at the RBOF and RRF. Since both of these tanks are located on chemical pad No. 1 and share a common flow path to waste tank EP38, the potential exists for large quantities of both chemicals to mix, if both tanks are breached at the same time. The mixing of nitric acid and sodium hydroxide produces sodium nitrate, which is routinely produced from process-related neutralization of the waste tank contents, water, and heat. Localized foaming and boilover could occur. General chemical handling safety rules recommend that bases and acids be stored apart so that if they simultaneously leak, they cannot mix. Event 347 or 345c in Table 3.3-14 provides a bounding estimate of consequence for this event. These events also have a radiological component. However, the consequences of the radiological component are bounded by the chemical consequences.

Sodium Dichromate and Phosphoric Acid

Specific amounts and concentrations of sodium dichromate and phosphoric acid were routinely mixed and used in solution to clean targets. (Target cleaning is no longer performed at the RBOF, and these chemicals are no longer used or stored at the RBOF). However, if at some future date target cleaning was resumed, the mixing of these two chemicals would not result in any hazardous reactions, and therefore no further analysis is required.

Sodium Nitrite and Aluminum Nitrate

Sodium nitrite and aluminum nitrate solutions are stored in tanks on chemical pad No. 2. No chemical reaction will occur as a result of their mixing. Therefore, no additional hazards are created, and no further analysis is required.

Oxalic Acid and Sodium Hydroxide

Oxalic acid (stored in Building 244-1H) and sodium hydroxide (stored on chemical pad No. 1) solutions are used for flushing filter EP14 to remove pluggage. Each of these chemicals is used separately to flush the filter and is subsequently removed from the filter and process piping by flushing. It is possible that mixing of the acid and caustic could occur, if standard operating procedures are not followed properly. According to the Handbook of Chemical Hazard Analysis Procedures (Ref. 26), mixing of oxalic acid and sodium hydroxide produces heat, which could occur if these chemicals are mixed. However, it is not anticipated that the heat generated would produce a release through boilover or foaming. Therefore, it is concluded that no additional hazards are created, and no further analysis is required.

Nitric Acid and Sodium Nitrite

Nitric acid is stored in tank EP50, which is located on chemical pad No. 1. Sodium nitrite is stored in a tank located on chemical pad No. 2. In the event of an earthquake or wind-induced failure of the rain shelter, both of these tanks or associated piping could leak or breach and release their contents onto their respective chemical pads. Chemical pad No. 1 gravity-drains to waste tank EP38. The chemical pad No. 2 sump is pumped to waste tank EP38. Given the extremely unlikely chance that the pump and controls survive the earthquake or rain shelter failure, the leaking sodium nitrite could be pumped to waste tank EP38 and mix with the nitric acid.

According to S-CLC-H-00203 (Ref. 25), the mixing of sodium nitrite and nitric acid is reactive and can generate a large volume of brown nitrous dioxide (NO_2) fumes. The temperature would also increase in waste tank EP38. Nitrous dioxide ERPG values are relatively low, 15 and 30 ppm for ERPG-2 and -3, respectively. Therefore, to reduce the risk associated with this scenario, a Plant Modification Traveler (PMT) (Ref. 38) was developed to modify the sump pump located

on chemical pad No. 2. This modification installed a disconnect switch on the pump motor power circuit and allowed de-energization of the pump motor under normal operating conditions. The PMT also modified the pump discharge piping to provide a means of isolating the pump discharge during normal conditions. Under unlikely conditions where the sodium nitrite tank leaks, motor power and pump discharge can be provided and the leak can be pumped to waste tank EP38. This capability will require verification of the contents of waste tank EP38 prior to pumping the spilled chemical.

Calculations have been performed to determine the maximum volume that the berm around chemical pad No. 2 can contain (Ref. 39). This volume is 963 gallons and will require the chemical inventory to be limited to 900 gallons as specified in Table 3.3-4. The berm around chemical pad No. 2 has also been qualified to withstand the collapse of the rain shelter without cracking (Ref. 40). The nitric acid and sodium nitrite hold tanks and the sodium nitrite dumpster have been qualified to all PC-2 seismic and wind loads (Ref. 41). Given the above administrative and design controls, the frequency of such an event is beyond extremely unlikely (Ref. 42).

Sodium Hydroxide and Sodium Nitrite

Sodium hydroxide is stored in tank EP52 on chemical pad No. 1, and sodium nitrite is stored in tanks on chemical pad No. 2. Both of these solutions could mix in waste tank EP38 due to the breach or spillage of their respective tanks. However, no chemical reaction will occur as a result of the mixing. Therefore, no additional hazards are created, and no further analysis is required.

Nitric Acid and Aluminum Nitrate

Nitric acid is stored in tank EP50, located on chemical pad No. 1. Aluminum nitrate is stored in a tank on chemical pad No. 2. In the event that both of these tanks leak or breach onto their respective chemical pads, the drainage from the pads will mix with waste in waste tank EP38. According to the Handbook of Chemical Hazard Analysis Procedures (Ref. 26), mixing of nitric acid and aluminum nitrate does not produce an additional hazardous substance or condition. Therefore, no additional hazards are created, and no additional analysis is required.

Sodium Hydroxide and Aluminum Nitrate

Sodium hydroxide is stored in tank EP52 on chemical pad No. 1, and aluminum nitrate is stored in tanks on chemical pad No. 2. Both of these solutions could mix in waste tank EP38 due to the breach or spillage of their respective tanks. According to the Handbook of Chemical Hazard Analysis Procedures (Ref. 26), mixing of sodium hydroxide and aluminum nitrate does not produce an additional hazardous substance or condition. Therefore, no additional hazards are created, and no additional analysis is required.

HAZARDS ANALYSIS ASSUMPTIONS

The following assumptions were made in conducting the hazards analysis:

- The frequency of occurrence for the majority of postulated events was taken from the Systems Analysis and/or Appendix A of S-CLC-H-00146 (Ref. 43). However, a few of the postulated events have not occurred at the RBOF and RRF. For these events, the conservative Rare Event Theory (Ref. 32) method was used to estimate initiator frequency of occurrence.
- A 1% release fraction, based on DOE-STD-1027-92, used to determine chemical source terms is adequate for use (Ref. 44).
- ORIGEN2.1 was used to calculate the RFA (worst-case fuel assembly radioactive material inventory) that the RBOF currently stores or is expected to store including domestic and foreign fuel types (Ref. 15 through 21). A 2-year decay (cooling period) was assumed (Ref. 22).
- The RBOF fissile material content is based on the fissile material activity of the RFA multiplied by 2,800 (the assumed number of possible fuel storage locations in the RBOF). The worst-case assembly is assumed to have a total of $1.14\text{E-}02$ Ci (or >5,500 grams) of U-235. The value is primarily based on the U-235 content of an irradiated RHF core.
- A basic fire protection program that controls the combustible loading in the RBOF and RRF is implicitly credited in minimizing the fire hazard.
- A basic worker competence level is maintained by a worker training and qualification program.
- Building ventilation filtration is not credited in mitigating any accident conditions but is considered as a possible source (e.g., source term contained in the exhaust filter itself) of radioactive material release.
- Waste tank EP37 contains no hazardous material.

The hazardous chemical compatibility evaluation included the assumption that the mixing of two hazardous chemicals could occur due to the following causes:

- Mixing of materials in process-related or interconnected drain piping or piping that leads to a common tank or sump
- Natural phenomena, causing storage vessels or containers in close proximity to breach
- Inadvertent human actions (i.e., wrong valve lineups or failure to follow process procedures)
- Equipment failure (i.e., malfunction, material failure, thermal stress, or corrosion)
- Inadequate process-related procedures

With the exception of the assumptions discussed above, assumptions specific to an event are discussed individually as appropriate.

3.3.2 HAZARDS ANALYSIS RESULTS

As discussed in Section 3.3.1, the hazards analysis consists of three basic analytical activities: hazard identification, hazard classification, and hazard evaluation. This section provides a discussion of the results of performing these activities.

3.3.2.1 Hazard Identification

The first step to ensure completeness of the identification of hazards is to list hazardous energy sources and hazardous material sources. As part of a systematic process to identify these sources for the RBOF and RRF, the following activities were performed:

- Applicable safety documentation was reviewed and evaluated to identify the hazardous energy sources and hazardous material sources associated with the RBOF and RRF. These documents included the following:
 - Existing Systems Analysis (Ref. 34)
 - SARs (Ref. 35, 45)
 - Basis for Interim Operation (BIO) (Ref. 46)
 - HADs (Ref. 13, 47)
 - Fire Hazards Analysis (Ref. 48)
 - Supporting engineering calculations and inter-office memoranda
- Selected operating procedures (i.e., fuel movement, cask handling, emergency operation procedure, and resin regeneration) for the RBOF and RRF were reviewed and evaluated.
- Available Preliminary Hazards Analyses (PHAs) and supporting documentation were reviewed and evaluated (Ref. 25, 43).
- Available Process Hazards Reviews and supporting documentation were reviewed and evaluated (Ref. 49, 50, 51).
- The SEPR database was reviewed and evaluated for the operational history of the RBOF and RRF (Ref. 52).
- Facility walkthroughs of the RBOF and RRF were performed, and numerous discussions with system experts, engineers, and operators were held.
- PHAs and PHA methodologies from other facilities or DOE sites (i.e., Central Laboratory Facility and Lawrence Livermore National Laboratory) were reviewed.

The results of the above-listed activities are documented in Tables 3.3-5 and 3.3-6 and identify the hazardous energy sources and hazardous material sources applicable to the RBOF and RRF.

Continuing with the systematic process that began with hazard identification and to ensure that all major hazards were evaluated and documented, the RBOF and RRF were divided into the following process areas and assigned a range of event numbers:

<u>Process Area</u>	<u>Event Number Range</u>
RBOF Basin Area	1-69
RRF Control Room, Chemical Pads, and Hose Racks	310-349
Waste Cell, Decontamination Cell, and RRF Cell	400-410
RBOF Control Room, Radiological Control and Health Physics (RC&HP) Office, Locker Room, Lunch Room, Electric Room, Roof Area, and Basement	450-460
Storage Building 244-1H	501-510

Facility segmentation for the hazard evaluation process ensures that each potential hazardous energy source and hazardous material source is evaluated for each facility segment, as applicable.

All hazardous energy sources and hazardous material sources were identified and documented in Tables 3.3-5 and 3.3-6. Radiological and chemical inventories were determined as discussed in Section 3.1 and are documented in Tables 3.3-2 and 3.3-3, respectively.

3.3.2.2 Hazard Classification

As stated in Section 3.3.1.2, the radiological hazard classification of each facility is determined by comparing the inventory of each radionuclide to the threshold quantities provided in DOE-STD-1027-92 (Ref. 4). The chemical hazard classification is determined by comparing the inventory of each chemical to the RQs of 40 CFR 302.4, the TPQs of 40 CFR 355, and the TQs of 29 CFR 1910.119 (Ref. 29, 30). The results of these comparisons are documented in the RBOF HAD (Ref. 13) and are summarized in the following sections.

RADIOLOGICAL RESULTS

The radiological analysis involved an evaluation of the maximum radionuclide inventory in the RBOF and RRF. The comparison of the RBOF/RRF radionuclide inventory, as shown in Table 3.3-2, to the threshold quantities set forth in DOE-STD-1027-92 (Ref. 4) and ANSI/ANS-8.1-1983 (Ref. 10) demonstrated the following:

- The actual inventories of some radionuclides (e.g., U-235, Pu-239) in the facility were greater than their Hazard Category 2 threshold limits.
- The sum of the ratios of the quantity of each radionuclide to the Hazard Category 2 threshold limit was greater than 1.

Therefore, based upon the radiological inventory (Table 3.3-2), the RBOF/RRF is categorized as a Hazard Category 2 facility.

CHEMICAL RESULTS

The RBOF/RRF chemical inventory was based on the maximum chemical inventory listed in Table 3.3-3. The following criteria were used in the initial screening of this inventory to eliminate certain chemicals from further evaluation in accordance with WSRC-MS-92-206 (Ref. 5):

- Any chemical with a maximum quantity of less than 1 pound is excluded consistent with the guidelines of 40 CFR 302 and 40 CFR 355 (Ref. 29).
- Any substance, to the extent it is used for personal, family, or household purposes, or is present in the same form and concentrations as a product packaged for distribution and use by the general public, can be eliminated from being categorized as a hazardous chemical in accordance with 40 CFR 355.20. This category would include such items as cleaners, paints, toners, diesel fuel, and petroleum jelly.
- The material is used in a laboratory setting, in laboratory-scale quantities. This category would include buffer solutions, resin, glove dust, indicator solutions, and silica gel. The exclusion of laboratory materials is consistent with 40 CFR 355.20, where the Environmental Protection Agency definition of "hazardous chemicals" specifically excludes "any substance to the extent it is used in a research laboratory or a hospital or other medical facility under the direct supervision of a technically qualified individual." "Laboratory" means a facility where "laboratory use of hazardous chemical" occurs. It is a workplace where relatively small quantities of hazardous chemicals are used on a nonproduction basis. "Laboratory scale" means work with substances in which the containers used for reactions, transfers, and other handling of substances are designed to be easily and safely manipulated by one person. "Laboratory scale" excludes those workplaces whose function is to produce commercial quantities of materials.
- Oil is excluded from the definition of hazardous material in accordance with DOE Order 5480.23 (Ref. 1).
- After reviewing the MSDS, a material is determined to be nonhazardous.

The remaining chemical inventory was compared to the RQs of 40 CFR 302.4, the TPQs of 40 CFR 355, and TQs of 29 CFR 1910.119. A hazard assessment was not required for any chemical that did not exceed the RQs, TPQs, or TQs. Following the initial screening process, the

chemicals were evaluated in accordance with the criteria of WSRC-MS-92-206. The chemical inventories that exceeded the RQs, TPQs, or TQs included 50% sodium hydroxide, 50% nitric acid, ethylene glycol, and 30% sodium nitrite. Therefore, these chemicals require a chemical hazards analysis to determine the potential consequences of an accidental release. Table 3.3-4 provides a comparison of chemical inventories to their respective RQs, TPQs, and TQs. Due to the concerns associated with the mixing of sodium nitrite and nitric acid, the inventory of sodium nitrite is limited to 900 gallons.

AMBERLITE IR-120 Plus (H) Resin and AMBERLITE IRA-400-I Resin do not have RQs, TPQs, or TQs. An alternative evaluation of the hazards associated with these chemicals was conducted by reviewing the MSDSs for AMBERLITE IR-120 Plus (H) Resin and AMBERLITE IRA-400-I Resin (Ref. 53, 54). According to the MSDSs, neither resin is considered a hazardous chemical under the OSHA Hazard Communication Standard, 29 CFR 1910.1200, Section 311/312 and Section 313. This chemical control law states that all components of these resins are nonhazardous. Both resin types fail to meet the characteristic definition of ignitability, corrosivity, or reactivity of the Resource Conservation and Recovery Act, and they are not listed in 40 CFR 261.33. Releases of these materials to air, land, or water are not reportable to the National Response Center under the Comprehensive Environmental Response, Compensation, and Liability Act or to state and local emergency planning committees under the Superfund Amendments and Reauthorization Act, Title III, Section 304. Based on the failure of AMBERLITE IR-120 Plus (H) Resin and AMBERLITE IRA-400-I Resin to meet any of the above hazardous chemical criteria, these resins may be considered nonhazardous. Thus, these resins do not require a chemical hazards analysis.

ERPGs, developed under the auspices of an American Industrial Hygiene Association technical committee, are estimates of concentration ranges for specific chemicals above which acute exposure would be expected to lead to adverse health effects of increasing severity for ERPG-1, -2, and -3. The chemical hazards analysis determines the maximum offsite, onsite, and local airborne concentrations resulting from a postulated release of the chemical inventory. In addition, these maximum airborne concentrations are compared to the applicable criteria (ERPG-3 or equivalent). A conservative analysis was performed for nitric acid, sodium hydroxide, and sodium nitrite in which it was assumed that the inventory was released within a 15-minute period.

The chemical inventories that exceeded the ERPG-3 values at 30 meters were 50% nitric acid and ethylene glycol. The chemical inventory that exceeded the ERPG-3 values at 100 meters was 50% nitric acid. No chemicals exceeded the ERPG-3 values at the site boundary.

Based upon these results, it was concluded that the facility is categorized as a Hazard Category 2 facility, as determined in accordance with WSRC-MS-92-206.

CRITICALITY RESULTS

The radiological inventory for the RBOF and RRF contains three primary fissile nuclides (U-233, U-235, and Pu-239) that present a potential criticality hazard for the facility. The radiological inventory of the RBOF and RRF is presented in Table 3.3-2. This amount of fissile material is well above the minimum critical mass limits specified in DOE-STD-1027-92 (Ref. 4). Therefore, in accordance with DOE-STD-1027-92, the RBOF/RRF must be categorized as a Hazard Category 2 facility.

3.3.2.3 Hazard Evaluation

The hazard evaluation is performed in accordance with the methodology provided in Section 3.3.1. The majority of the information required to perform the hazard evaluation was obtained through system-by-system reviews of facility documentation, facility walkthroughs, and discussions with facility personnel. The RBOF and RRF were segmented into process areas (defined in Section 3.3.2.1), and each process was located within the process areas and was analyzed against the event categories (E-1 through E-10), one at a time, postulating event scenarios for the applicable hazardous energy sources and hazardous material sources listed in Tables 3.3-5 and 3.3-6. The identified event scenarios cover the spectrum of possible events for a given hazard from small consequence events, for which existing procedures or equipment is acknowledged to provide adequate protection, to reasonable worst-case conditions, in terms of both event frequency and consequence magnitude.

The results of these activities are documented in Tables 3.3-13 through 3.3-17, which identify all events applicable to the RBOF and RRF and provide the results of a hazard evaluation (risk rank) of each event.

This evaluation process resulted in the identification of 136 potential events for the MOI and onsite worker involving hazards and/or hazardous materials that were evaluated according to the methodology discussed in Section 3.3.1. The risk rank for each postulated event is determined by locating the event on the risk matrix (Table 3.3-10), based upon the consequence level and frequency level of the event. This risk rank is provided in Tables 3.3-13 through 3.3-17. Table 3.3-11 is a summary table of all events and their respective risk rank.

None of the 136 events identified have a risk rank of 1 or 2. Six of the 136 potential events identified in Tables 3.3-13 through 3.3-17 with a risk rank of 4 (Events 51, 310, 314a, 316, and 318) require further quantitative analysis. Forty-seven events with a risk rank of 3, 5, or 7 require a bounding analysis for each event type. Forty-six events with a risk rank of 6 or 8 through 13 will require no further analysis. Thirty-five events involve standard industrial hazards and received a risk ranking but are not considered for further analysis in accordance with DOE-STD-3009-94 (Ref. 2). Table 3.3-11 identifies the standard industrial hazard events. In addition, NPH events that do not result in the release of radiological or hazardous material outside of the facility do not require further evaluation (Ref. 2, 55).

A comparison between the event categories in the hazard evaluation tables and the event categories of WSRC Procedure Manual 11Q (Ref. 3) is documented in Table 3.3-12. The results of this review ensured that each of the events listed in Tables 3.3-13 through 3.3-17 was evaluated to determine that all applicable NPH events, externally induced events, and process-related events were covered in accordance with the new guidance of WSRC-TR-94-0466.

The following sections provide a brief discussion of the accident scenarios identified and documented in Tables 3.3-13 through 3.3-17.

E-1: FIRE/EXPLOSION

As noted in Tables 3.3-5 and 3.3-6, the RBOF and RRF contain hazardous energy sources capable of exploding or igniting the explosive and combustible material associated with the facilities' structures and operations. Therefore, events involving these hazards were postulated for each of the evaluation areas established in Section 3.3.2.1.

Events 1, 2, 2a, 3a, 311, 312, 450, 451, and 501

A Fire Hazards Analysis for the RBOF and RRF (Ref. 48) has concluded that a fire within the RBOF complex will pose no undue risk to the public or site workers. This conclusion is attributed to the noncombustible construction, lack of continuity of combustibles in most areas, and the nature of the operation within the facility. Adjacent fires and explosions were eliminated from consideration because nearby facilities are far enough away so as not to impact this area.

As part of the mission for the RBOF, fuels are routinely received or shipped from/to the reactor disassembly basins or the canyons using the 70-ton cask. Offsite fuels are shipped to the RBOF in casks certified by the Department of Transportation (DOT), the Nuclear Regulatory Commission (NRC) or the International Atomic Energy Agency (IAEA). The hazards analysis identified a potential scenario involving these types of casks during shipment of uranium-aluminum (U-Al) fuels. It is possible for residual water left in the cask to decompose due to radiolysis resulting in hydrogen gas production. Separations Incident SI-81-9-96 (Ref. 57) documents such an event where a minor explosion occurred during venting of a High Flux Isotope Reactor (HFIR) cask.

In addition, water can react with metallic uranium to form uranium oxide and hydrogen gas. Some fuels in the RBOF were shipped to the RBOF with failed fuel cladding (TRR & EBR-II) further increasing the likelihood of hydrogen gas generation (Ref. 58). The hydrogen gas generation is not a concern during storage in the RBOF (see Event 11) due to the small amounts of gas generated and the large volume of the basin area. However, during transportation of known failed fuel in the 70-ton cask, precautions must be taken to ensure that the lower flammability limits are not exceeded by limiting the amount of time the fuel remains in the cask

without the cask being vented or unloaded in accordance with the Fuel Receipt and Shipping Program.

The health and safety consequences of postulated fires/explosions within the RBOF (Events 1, 2, 3a, 311, 312, 450, 451, and 501) are defined qualitatively based on the potential for physical injury to the worker. These postulated events have the potential for injury to facility workers, and are primarily a standard industrial hazard. These postulated fire events release little or no radiological or chemical inventories due to the small hazardous inventories involved, or in the case of Event 2a, the shipping cask prevents the release of hazardous inventories. Therefore, the consequence level for these events is low based on the consequence criteria in Table 3.3-8, and the hazard evaluation tables were annotated accordingly.

Event 3

Event 3 involves a minor explosion during venting (through the building exhaust system) of an offsite or onsite cask. Any gaseous radioactivity in the cask, which is routinely released during cask venting operations, is small and does not present a significant radiological hazard to the facility workers. Additionally, even if the explosion causes significant fuel damage, only the volatile fission products would be released. In the discussion for Events 7, 10, and 11, it is demonstrated that if all the fuel assemblies in the basin were damaged, it would still only result in low consequences to the onsite and offsite receptors.

Events 313 and 314

Events 313 and 314 have the potential to release chemicals in excess of the ERPG-2 values and are moderate consequence events.

Event 400

As noted in past analysis (Ref. 34), a fire and explosion could occur in waste tank EP38 (Event 400) in the event that the coolant (containing a flammable antifreeze) of a received cask, when discharged to the waste tank, results in a flammable or explosive concentration of vapors in the tank. Nitric acid, albeit diluted (typical waste volumes from regenerating resin are 4,000 gallons of which less than 4% is nitric acid [Ref. 59]), may be present in waste tank EP38 and can exacerbate this problem. Nitric acid, when mixed with antifreeze such as ethylene glycol (in sufficient concentrations), can produce an exothermic reaction, which could produce flammable reaction products or mixtures.

There is no evidence that a cask containing an antifreeze fuel-coolant has ever been received at the RBOF. However, the potential exists because Appendix B to the offsite cask shipping contract states "the use of any liquid coolant other than water as the primary cooling medium, or the use of additives, such as anti-freeze, will be considered on an individual basis prior to shipment and may require special handling." There are no provisions that allow for the use of

such additives in onsite casks under any circumstances; therefore, this is only a concern with offsite shipments.

The Nuclear Assurance Corporation (NAC) Legal Weight Truck (LWT) cask is a spent fuel shipping cask used to transport Pressurized Water Reactor (PWR), Boiling Water Reactor (BWR), and Material Test Reactor (MTR) fuel. The NAC LWT cask has a closed neutron shield tank that contains a water/ethylene glycol mixture. However, the NAC LWT shield tank is infrequently drained and is not discharged to the waste tank.

For the event to occur, the facility must receive a cask containing antifreeze, the contents of the cask must be discharged to the waste tank, and an ignition source must exist in the waste tank. The frequency of the facility receiving casks containing antifreeze is estimated using the Rare Event Theory (3 occurrences/32 years, or $9.38\text{E-}02/\text{yr}$). The Fuel Receipt and Shipping Program prohibits the transfer of cask contents containing flammable antifreeze to the waste tank. Therefore, discharging the cask contents to the waste tank would violate administrative controls, which is estimated to occur at a probability of $5.0\text{E-}03$ (Ref. 60). The probability of a spark source existing in the waste tank is estimated from a review of three recent ignition source studies performed for H-Area waste tanks and the Defense Waste Processing Facility (Ref. 61, 62, 63). The largest probability developed in these three studies (0.09) is assumed to bound the probability of ignition in waste tank EP38. Therefore, the overall probability of this event occurring is estimated to be $9.38\text{E-}02/\text{yr} * 5.0\text{E-}03 * 0.09 = 4.22\text{E-}05/\text{yr}$, which is an extremely unlikely event.

For the event to occur with the NAC LWT cask, the cask shield tank must be drained at the RBOF, the contents of the drain tank must be discharged to the waste tank, and an ignition source must exist in the waste tank. The frequency of the facility draining the NAC LWT shield tank in the RBOF facility is estimated using the periodicity of draining the shield tank and a conditional probability that the NAC LWT is in the RBOF facility (once every 5 years times a conditional probability of 0.5, or $0.1/\text{yr}$). The Fuel Receipt and Shipping Program prohibits the transfer of cask contents containing flammable antifreeze to the waste tank. Therefore, discharging the cask contents to the waste tank would violate administrative controls, which is estimated to occur at a probability of $5.0\text{E-}03$ (Ref. 60). The probability of a spark source existing in the waste tank is estimated as discussed above to be 0.09. The probability of this event occurring is estimated to be $0.1/\text{yr} * 5.0\text{E-}03 * 0.09 = 4.5\text{E-}05/\text{yr}$. The overall frequency of this event occurring is estimated by adding the two frequencies, which is $8.7\text{E-}05/\text{yr}$ (an extremely unlikely event).

Based on S-CLC-H-00237 (Ref. 64), an explosion in waste tank EP38 results in dose of 0.07 mrem to the MOI. Comparing these doses to consequence criteria in Table 3.3-8, the event would be classified as a low consequence event. However, since a life-threatening injury to the facility worker is possible due to the tank explosion, the event was annotated as moderate consequence.

Event 400a

During regeneration of cation resin from an RBOF deionizer or from a reactor basin portable deionizer, an overpressurization could occur in the resin regeneration tank (Event 400a) when undiluted nitric acid (6 molar or greater) is added to the cation resin being regenerated. Based on calculations documented in S-CLC-H-00203 (Ref. 25), it was assumed that a bounding case explosion occurs in the regeneration tank resulting in a release into the shielded waste cell and subsequently to the environment. The resulting dose was 19.3 mrem to the individual at the Onsite Evaluation Point (OEP) and 2.39 mrem to the MOI, which is a low consequence event as defined in Table 3.3-8.

Events 310 and 400b

Due to the potential radioactivity of the 100-Area portable deionizer resin and the potential flammable source of fuel in the truck used to transport the deionizer to the RRF, a fire in the portable deionizer has the potential to expose the onsite worker to greater than 25 rem. For the RBOF deionizer fire scenario, the lack of an ignition or flammable source in the decontamination cell reduces the frequency of this event to extremely unlikely. The only identified mechanism for heating the resin to flammability limits is by alpha emission of the resin. S-CLC-H-00368 (Ref. 65) determined that the maximum loading of the RBOF deionizer would not contain sufficient amounts of alpha-emitting isotopes to heat the resin to the point of self-ignition. Tables 3.3-14 and 3.3-15 were annotated accordingly.

E-2: LOSS OF CONTAINMENT

As noted in Table 3.3-2, the operation of the RBOF and RRF involves hazardous radiological materials. The RBOF/RRF basins, process systems, vessels, fuel and target cladding, and damaged fuel storage canisters provide the primary containment of these materials. Natural phenomena, extrinsic hazards, and hazardous energy sources (Table 3.3-5) have the potential for damaging the primary containment allowing loss of material. Therefore, events involving these hazards were postulated for each of the evaluation areas established in Section 3.3.2.3.

Reviewing Table 3.3-13, it can be seen that loss of containment can result in contamination (worker and/or equipment) and/or release (airborne or subsoil) of activity. The two types of releases associated with loss of containment events are (1) airborne release of particulate and gaseous material followed by atmospheric transport to man, and (2) subsurface release of activity followed by migration through the soil before release to surface streams.

Surface Contamination

Events 4 and 23

Casks as received are sometimes damaged or deficient (Events 4 and 23), and these deficiencies can compromise the integrity of the cask and result in the spread of contamination. Once the cask has been received, this spread of contamination only impacts the interior of the RBOF facility. Therefore, event consequence is based on a qualitative assessment of the potential for occupational illness or prompt death to the facility worker.

The present DOT limits for offsite cask surface contamination at the beginning of transportation are 220 dpm/100 cm² for alpha contamination and 2,200 dpm/100 cm² for mixed beta-gamma contamination. DOT makes allowances for possible leaching during transportation allowing 10 times (i.e., 2,200 and 22,000 dpm/100 cm², respectively) the above values during and after transport. SRS onsite cask contamination has not posed a hazard to personnel in that removable contamination levels are typically in the range of 4,000 to 40,000 counts per minute (40,000 to 400,000 dpm) of mixed beta-gamma radiation per 100 cm² of area. Therefore, assuming a reasonable worst-case contamination level of 400,000 dpm/100 cm² (< 1 Ci), it was concluded that this event resulted in low consequence to the facility worker. Table 3.3-13 was annotated accordingly.

Event 9

As noted in past analysis (Ref. 34), there have been occasions when the RBOF basins have overflowed, contaminating the immediate area. Most overflows involve minor "splashing" of basin water and pose no consequence to facility workers. In assuming a worst-case scenario for the volume of water overflowing the basin, past facility analysis documents that an increase of 20,000 gallons of water in the basin did not result in an overflow but did result in the high basin overflow level to discharge to the seepage basins. Therefore, a 20,000-gallon (75,600 liters) overflow was assumed for the worst-case scenario.

Event consequence is based on a qualitative assessment of the potential for occupational illness or prompt death to the facility worker. As a result of fuel handling and cropping operations in which the fuel was inadvertently exposed, the maximum activity of the isolated basin water was approximately 0.1 $\mu\text{Ci/mL}$ (Ref. 34). Assuming a reasonable worst-case water activity level of 0.1 $\mu\text{Ci/mL}$ and an overflow of 75,600 liters, the total curies involved in the overflow would be 7.6 Ci, which results in negligible consequences to the facility worker when compared to the consequence criteria in Table 3.3-8. Table 3.3-13 was annotated accordingly.

Subsurface Releases

Events 8, 321, 406, and 407

As noted in past analysis (Ref. 34), the potential exists for accidental releases of liquid waste. Events 8, 321, 406, and 407 involve the subsurface releases of liquid radioactive waste. Consequences of releases to the soil are considered negligible because the travel time to reach a surface stream allows for most of the activity to decay, making the consequence negligible.

To address leak detection concerns identified in RRD-RSE-940259 (Ref. 66), WSRC has evaluated the results from the West Valley (lead site for this area of technology development) programs for applicability to the RBOF. As a result of this review, improvements in level detection have been made by modifying the RBOF level alarm circuits, which have enhanced the response characteristics of the system.

Releases to the subsoil are negligible based on the analysis in SRT-EES-93-0143 (Ref. 67). This analysis estimates the offsite individual exposure resulting from a hypothetical release of 1.3 million gallons of high-level waste over a 24-hour period from waste storage tanks in H Area. Based on this analysis, the largest radionuclide source term is Cs-137 ($4.8\text{E}+07$ Ci), with tritium ($1.2\text{E}+02$ Ci) and Pu-240 ($2.66\text{E}+03$) also being present. The analysis considers migration of the nuclides to the water table in the subsoil below the tanks without credit for the time it would take to travel through the unsaturated soil underneath the tank. The nuclides flow in the groundwater to the nearest discharge point of the Four Mile Branch. The results of this analysis indicate that the resulting maximum offsite individual exposure is 1.3 mrem. Due to the transport time through the subsoil, this exposure is estimated to occur 50 years after the release.

A total release of RBOF basin water and waste tank contents as a result of a major leak into the subsoil is bounded by this analysis. The analysis parameters can be directly applied to the RBOF with the following built-in conservatisms:

- The RBOF basin water contains approximately 189 Ci total activity (assuming a basin volume of 500,000 gallons and a basin activity of $0.1 \mu\text{Ci/mL}$). Orders of magnitude are less than those of the waste tanks. The release of the total basin water content would likewise result in much lower individual exposure consequences.
- The conservatisms built into the analysis also apply to the RBOF. In fact, RBOF subsoil releases would have to travel longer distances to Four Mile Branch or Upper Three Runs Creek, resulting in lower exposures.

Therefore, dose estimates from this pathway are not estimated. On this basis, it was concluded that postulated Events 8, 321, 406, and 407 were negligible consequence level events as defined by Table 3.3-8. Table 3.3-13 was annotated accordingly.

Events 8a, 8b and 8c

Event 8a evaluated the impact of a low water level in the storage basin on the stored fuels due to a leak in an operating filter deionizer (FD) system. As part of the analysis of Event 8a, consideration was given to the impact that a worst-case basin draindown would have on the fuels presently stored in the basin. This event assumes that no actions are taken to add water to the basins and that the water level is reduced to the point of exposing the stored fuel. An analysis of the MURR fuel (Ref. 68) was performed to address the potential impact of the basin draindown and impact on the presently stored fuel assemblies. This analysis determined that the worst-case fuel assembly (MURR) with a decay heat of 135 watts would not melt (i.e., beyond extremely unlikely) in a normal storage configuration as a result of a basin draindown. If the fuel had a higher decay heat than the values assumed for the MURR fuel, it would be possible for the fuel to melt in a basin draindown condition and there would be offsite consequences, which would rank as a high consequence event. Table 3.3-13 was annotated accordingly for Event 8a.

Events 8b and 8c considered the effect on the facility worker of exposed fuel as a result of a reduced basin water level. For event 8b, the cause of the reduced basin level is a filter-deionizer system leak, which if undetected could result in exposing some of the longer fuel assemblies presently stored in the RBOF. At this water level, the basin pumps would cavitate, and thus the loss of basin water through the filter-deionizer system would terminate. Chapter 2 provides additional details on the elevation at which the basin pumps are located. For event 8c, the cause of the reduced water level is a basin leak caused by a dropped shipping cask. In this event, the shipping cask is dropped while either moving it from the trailer or railroad car to the cask wash pit or from the cask wash pit to the cask basin. The impact from the dropped shipping cask could result a crack in the basin walls and subsequent leakage of basin water to the environment. The RBOF basins were not specifically analyzed for the size or location of the crack as a result of the dropped shipping cask. To address this concern, basin seepage analysis was performed to determine the rate at which the cask or storage basins would drain with various assumed crack sizes. The RBOF Basin Seepage Analysis (K-CLC-H-00110, Ref. 222) determined the time required to lower the basin water level such that the top of the tallest fuel assemblies would be exposed or a basin draindown of 4-1/2 feet below the full basin level. The basin cracks were assumed to be of constant width and either run longitudinally or transversely along the bottom of the basin and up its sides. From this analysis, the time required to lower the water level was determined to be approximately 400 hours for a crack width of 48 inches and 648 hours for a crack width of 2-1/2 inches. This would allow sufficient time for an integrated facility response to a basin leak.

Assuming the worst-case exposure based on the RFA, basin draindown events could result in a potential high radiation exposure to the facility worker. Maintaining the basin water level during manned operation of the facility was credited in preventing the exposure of fuel and the resulting high radiation dose to the facility worker. Assuming failure of this control during the operation or standby mode, these events would cause an area radiation alarm to be activated and the facility workers to immediately evacuate the area, thereby mitigating the consequence.

During unmanned operations, the operability of the Nuclear Incident Monitors (NIMs) and associated remote alarms is credited in providing an indication of high radiation conditions to a remote area (Building 221-H Control Room). This indication prevents an inadvertent re-entry into the RBOF under the extremely unlikely condition of a reduced basin water level and the inoperability of the NIMs. Escort of operations personnel by Radiological Control Operations in re-manning the facility during this condition was credited in mitigating the consequence of this event. Therefore, it is assumed that these events result in a low consequence to the facility worker, and Table 3.3-13 was annotated accordingly. It is assumed that if these events cause an area radiation alarm to become activated during the operation or standby mode, the facility workers would immediately evacuate the area, thereby mitigating the consequence.

The subsurface release portion of Events 8a, 8b and 8c is bounded by the analysis discussed above. The subsurface release would have negligible consequences because of the long travel time during which most of the activity decays.

Airborne Release

The quantity of radionuclides at risk for events involving spent fuel is based on the RFA cooled for 2 years. Of all the fuels presently stored or expected to be stored in the RBOF basins, the RFA is bounding and is therefore used as the reference fuel when evaluating consequences involving irradiated fuel assemblies. The quantity of chemicals at risk for events is based on the chemical inventory data for the RBOF HAD (Ref. 13).

Events 5, 6, 12, and 13

Fuel as received is sometimes breached or damaged (Ref. 34). Fuel breach could result in the release of volatile fission products to the shipping cask. Subsequent cask handling operations (Events 5, 6, 12, and 13) could result in the release of radioactive gases from the cask to the RBOF and/or the environment via the RBOF building exhaust filters and exhaust stack. These offsite shipping casks are qualified in accordance with 10 CFR 71 (Ref. 31) and 49 CFR 173 (Ref. 69). This qualification establishes radiological consequence limits for the shipping casks for hypothetical accident conditions, including a release limit of 10,000 Ci of Kr-85 within a 1-week period and an external radiation dose rate limit of 1 rem/hr at 1 meter from the external surface of the shipping cask. Based on an ORIGEN computer code analysis of the worst-case fuel assembly (RFA), the assembly contains 1,050 Ci of Kr-85 (Ref. 21). A conservative worst-case assumption would be that only 30% of the Kr-85 contained within the affected fuel assemblies would escape (Ref. 70).

Regulatory Guide 1.25 (Ref. 70) provides release fractions for commercial type fuels (BWR and PWR), which are comprised of ceramic oxide pellets in zircaloy cladding with a gap between the two. This gap is usually pressurized, which would result in higher release fractions when compared to those experienced from a breach of a solid matrix type fuel (U-Al) such as those found in the RBOF. Comparing this activity and the limits established by the 10 CFR 71

qualifications to the minimum radiological source terms required to meet consequence level threshold values results in a determination of low consequences to the facility worker and the public as defined in Table 3.3-8 (Ref. 25).

Event 5a

WSRC-RP-89-715 (Ref. 71) previously identified the bounding source terms for the SRS cask during transportation of radioactive materials at SRS. These source terms are listed in the hazards analysis (Ref. 25). WSRC-RP-89-715 analyzed two different source term release events applicable to the RBOF. Two shipment descriptions are specified in Table I-1 of WSRC-RP-89-715: (1) irradiated fuel from 100 and 200 Areas, and (2) sections of fuel from the RBOF to Building 773-A. For conservatism, these two events were combined with a worst-case release of $2.00\text{E}-5$ Ci of Pu-239, 10 Ci of beta-gamma, and $3.0\text{E}+00$ Ci of H-3.

The appropriate Airborne Release Fractions (ARFs) of $1.0\text{E}-04$ (Pu-239) and $1.0\text{E}+00$ (for the remaining contributors) were applied to the total curies of activity, yielding the expected source term for postulated events involving onsite cask handling operations that result in airborne releases. Comparing this activity to the minimum source terms required to meet consequence level threshold values, it was concluded that this event results in low consequences to the facility worker and the public as defined in Table 3.3-8. Table 3.3-13 was annotated accordingly (Ref. 25). The SRS 70 ton cask is presently not qualified by 10 CFR 71, and therefore the values of 10 CFR 71 cannot be assumed.

Events 7, 10, and 11

During fuel handling operations and storage, the potential exists for fuel cladding to be damaged, to be cut, or to breach (Events 7, 10, and 11), resulting in the release of radioactive material and gases to the RBOF basin. Assuming the isotopic inventory of an RFA and the release fractions in accordance with Regulatory Guide 1.25 (Ref. 70), damage to fuel results in the release of 30% of the Kr-85. A bounding analysis is provided by assuming that all 2,800 RFAs are damaged concurrently, each containing 1,050 Ci of Kr-85. (No other noble gases are present in the isotopic inventory as a result of the 2-year cooling period). A comparison of the total radiological release to the minimum radiological source terms in S-CLC-H-00203 (Ref. 25) results in a determination of low consequences to the facility worker and the public as defined in Table 3.3-8. Table 3.3-13 was annotated accordingly.

Event 10a

Inadvertent cutting of the target core (Event 10a) can also cause a release of airborne activity to the atmosphere via the RBOF stack. The cutting of Mark VI-B targets in the basin would result or has resulted in releases of up to 65 Ci of tritium (Ref. 35). Total release of vapors containing volatile forms of materials (e.g., iodine, irradiated water) upon loss of containment is assumed. Therefore, since tritium is involved, an ARF of $1.0\text{E}+00$ is applied to the total curies of activity

listed above, yielding the expected source term from inadvertent cutting of the target core. Comparing this activity (65 Ci) to the minimum source terms required to meet consequence level threshold values, it was concluded that this event results in low consequences to the public and facility worker as defined in Table 3.3-8. The target cutting events of Table 3.3-13 were annotated accordingly.

Events 7a and 11a

During target handling operations and storage, the potential exists for target assemblies to be damaged or to breach (Events 7a and 11a) resulting in the release of tritium to the RBOF basin. Using a Mark VI-B target as a reference target for this analysis, these events would result in releases comparable to target cutting. On this basis, it was concluded that these events result in low consequences to the public and facility worker as defined in Table 3.3-8. Table 3.3-13 was annotated accordingly.

Event 401

Breach of waste tank EP38 (Event 401) could occur resulting in the loss of liquid waste to the shielded waste cell. The activity involved in this postulated event would be the same as the radioactive isotope loading on the deionizer involved in the nitric acid reaction (Event 400a) previously discussed. However, the ARF associated with the nitric acid reaction event would be larger due to the available energy involved with the chemical reaction as compared to the assumed standing pool of spilled liquid. Therefore, the amount of activity assumed to become airborne in the waste cell and released to the environment would be less than the nitric acid event, and the consequences associated with this event would be bounded. Therefore, this event was assigned a negligible consequence level as defined in Table 3.3-8.

Events 314a, 314b, 315, 316, 317, and 318

The RRF uses 50% sodium hydroxide and 50% nitric acid to regenerate anion and cation resin, respectively. According to the chemical inventory, 1,000 gallons of each chemical are stored on chemical pad No. 1 (Ref. 13). Additionally, the capability exists to store 2,600 gallons of 30% sodium nitrite on chemical pad No. 2 in two tanks. However, due to concerns with the mixing of sodium nitrite and nitric acid, the allowable inventory of sodium nitrite is 900 gallons. These chemicals were evaluated by the RBOF/RRF hazard classification and found to be chemicals requiring hazard assessment (maximum inventory is a quantity not generally accepted by the public) (Ref. 13). The assessment documented in S-CLC-H-00203 (Ref. 25) determined that for each chemical an unmitigated release of the maximum inventory would not exceed the PEL-TWA values for the MOI. However, the hazard classification determined that the same release would exceed the ERPG-2 and -3 values for the facility worker. Therefore, only consequences to the facility worker from events involving the release of these chemicals were evaluated.

If these tanks are damaged by extrinsic or NPH events that remove the tanks from the chemical pad such that the spilled contents were not contained within the berm around the chemical pads, there could be potential significant consequences to the facility worker. On this basis, the following conclusions were made:

- Releases involving greater than 72 gallons of 50% sodium hydroxide (Event 316), 71 gallons of 50% nitric acid (Event 318), or 167 gallons of 30% sodium nitrite (Event 314a) are moderate consequence events.
- Releases involving less than 72 gallons of 50% sodium hydroxide (Event 315) or less than 71 gallons of 50% nitric acid (Event 317) are low consequence events.
- Releases involving less than 167 gallons of 30% sodium nitrite (Event 314b) are low consequence events.

Table 3.3-14 was annotated accordingly.

Event 328c

Ethylene glycol is added to the RBOF facility Heating, Ventilation, and Air Conditioning (HVAC) system, as needed, to provide freeze protection and can be used as a shipping cask cooling medium. To determine whether or not releases of this chemical would result in moderate consequences to the facility worker, the minimum source term necessary to meet the moderate consequence level threshold value for the facility worker was determined (Ref. 25). The amount of released ethylene glycol, following loss of containment, necessary to exceed the moderate consequence level threshold values is 123 gallons (Ref. 25). However, the Environmental Protection Agency's Open Dump Model, which accounts for the spilled chemical's physical properties (e.g., vapor pressure), results in a much lower release fraction for ethylene glycol, approximately 0.004% (Ref. 72) versus the 1% used in the RBOF HAD (Ref. 13). Therefore, a more accurate estimate of the quantity of ethylene glycol needed to exceed the moderate consequence level threshold value, ERPG-2, is 1% divided by 0.004% times 123 gallons. The resultant quantity is 30,750 gallons. On this basis, it was concluded that Event 328c involving the release of this chemical results in low to negligible consequences to the facility worker as defined in Table 3.3-9.

Events 324 and 334

Breach of portable deionizers or associated piping, hoses, and valves (Events 324 and 334) while at the RRF could occur, resulting in the release of contaminated water to the immediate area. The activity involved in these postulated events is mainly due to Cs-137. Thus, for a reasonable worst-case assumption, there would be a 50.0-Ci release of activity based on the typical curie content of a basin deionizer (Ref. 59). Comparing this activity to the minimum source terms required to meet consequence level threshold values as defined in Table 3.3-8, it was concluded that these events result in low consequences.

Event 401a

Nitric acid is stored in tank EP50, which is located on chemical pad No. 1. Sodium nitrite is stored in two tanks located on chemical pad No. 2. In the event of an earthquake or wind-induced failure of the rain shelter, these tanks or associated piping could leak or breach and release their contents onto their respective chemical pads. Chemical pad No. 1 gravity-drains to waste tank EP38. Chemical pad No. 2 sump is pumped to waste tank EP38. Given the extremely unlikely chance that the pump and controls survive the earthquake or rain shelter failure, the leaking sodium nitrite could be pumped to waste tank EP38 and mix with the nitric acid. According to S-CLC-H-00203 (Ref. 25), the mixing of sodium nitrite and nitric acid is reactive and can generate a large volume of brown nitrous dioxide (NO_2) fumes. The temperature would also increase in waste tank EP38 (Ref. 25). Nitrous dioxide ERPG values are relatively low, 15 and 30 ppm for ERPG-2 and -3, respectively.

Therefore, to reduce the risk associated with this scenario, a PMT (Ref. 38) was developed to modify the sump pump located on chemical pad No. 2. This modification installed a disconnect switch on the pump motor power circuit and allowed de-energization of the pump motor under normal operating conditions. The PMT also modified the pump discharge piping to provide a means of isolating the pump discharge during normal conditions. Under unlikely conditions where the sodium nitrite tank leaks, motor power and pump discharge can be provided and the leak can be pumped to waste tank EP38. This will require verification of the contents of waste tank EP38 prior to pumping the spilled chemical.

Calculations have been performed to determine the maximum volume that the berm around chemical pad No. 2 can contain (Ref. 39). This volume is 963 gallons and will require the chemical inventory to be limited to 900 gallons as specified in Table 3.3-4. The berm around chemical pad No. 2 has also been qualified to withstand both the collapse of the rain shelter without cracking and PC-2 seismic loads (Ref. 40, 73). The nitric acid tank, sodium nitrite hold tank and sodium nitrite dumpster have been qualified to all PC-2 seismic and wind loads (Ref. 41). Given the above administrative and design controls, the frequency of such an event is beyond extremely unlikely (Ref. 42). Table 3.3-15 identified this event as Event 401a.

Event 402

This event postulates a breach of the low-level waste trailer or other containers (e.g., 55-gallon drums) used to contain the liquid from the cask wash pit. Waste trailers or drums may be used in lieu of waste tank EP38. The maximum activity associated with the waste trailer or drum inventories in the RBOF is assumed to be bounded by the 100-Area unshielded trailer inventories (100-Area basin sludge) analyzed in WSRC-RP-89-715 (Ref. 71). Assuming the entire contents of the trailer or drums is spilled and comparing this activity to the minimum source terms required to meet consequence level threshold values, it was concluded that these events result in low consequences.

E-3: LOSS OF CONFINEMENT

Table 3.3-6 identifies hazardous material associated with the operation of the RBOF and RRF. While the fuel cladding, RBOF basins, process systems, and vessels provide primary and secondary containment of these hazardous materials, the facility itself, along with the ventilation system, provides for the confinement and/or filtered release of any material lost from the primary and secondary containments. Natural phenomena, extrinsic hazards, and hazardous energy sources have the potential for damaging this confinement, allowing an unfiltered release of material to the environment. Therefore, events involving these hazards were postulated for each of the evaluation areas established in Section 3.3.2.1.

Events 16 and 17

As noted in past analysis (Ref. 34) and Appendix A of S-CLC-H-00146 (Ref. 43), on occasion the RBOF carport door has been damaged by tractor trailers, and on other occasions the RBOF carport door has been inoperable for various reasons. When this occurs, while unlikely with building exhaust fans running, there is the potential for an unfiltered ground release to the atmosphere. Under normal operating conditions, the amount of activity being exhausted through the building filters and stack is negligible. However, during operations such as fuel cropping or target cropping, based on the RFA (worst-case) and referenced target assembly (Mark VI-B) for analysis purposes, there is the potential of releasing fission product gases contained within the assemblies. A reasonable worst-case assumption would be bounded by the concurrent damage to all 2,800 assemblies discussed previously (Events 7, 10, and 11). Comparing this activity to the minimum source terms required to meet consequence level threshold values as defined in Table 3.3-8, these events result in low consequences to the facility worker and the public.

Events 18 and 19

In addition to an inoperable carport door, unfiltered ground releases can be caused by NPH events (Events 18 and 19) that damage the RBOF structurally. Under these conditions, a reasonable worst-case assumption would be that the amount of material released is the same as that released via the carport door. Furthermore, the same assumptions, ARF, and source term apply. (Refer to event category E-10 for further discussion relative to NPH-induced events.)

Events 20 and 21

Damage, failure, or pluggage of the RBOF building exhaust filters could result in minor releases of radioactive particles. Historical swipe surveys of removed building exhaust filters by Radiological Control Operations have resulted in maximum readings of approximately 500,000 dpm/100 cm² (<1 Ci). Assuming the building exhaust filter fails with a reasonable worst-case release of 100 Ci of Cs-137 and without applying release fractions (for conservatism), this results in a total release of 100 Ci of Cs-137. Comparing this source term to the minimum source term

necessary to meet consequence level threshold values as defined in S-CLC-H-00203 (Ref. 25), it was concluded that these events result in negligible consequences to the facility worker and the public.

Event 329c

The consequences from tornado-driven missiles penetrating the RRF shielded wall or the roof and entering the resin regeneration cell are bounded by Event 400a (explosion in the resin regeneration tank). The dose from the explosion in the resin regeneration tank was 19.3 mrem to the individual at the OEP and 2.39 mrem to the MOI. Comparing these doses to Table 3.3-8, Table 3.3-14 was annotated as low consequence.

E-4: INJURY TO WORKERS

The RBOF and RRF contain hazardous energy sources and hazardous material sources capable of causing injury to the workers during normal routine operations. Therefore, events involving these hazards were postulated for each of the four evaluation areas established in Section 3.3.2.1. Reviewing Tables 3.3-13 through 3.3-17, it can be seen that these E-4 events can be grouped into three types of events that can cause injury to workers: exposure to radiation, exposure to hazardous chemicals, and industrial events.

Events 26, 26a, 27, 28, 29, 30, 31, 32, 33, 33a, 33b, 61, 330, 331, 332, 333, and 335

For postulated E-4 events considered industrial in nature, where radiological or toxicological hazards are not involved, the consequences were based on the potential for physical injury to the facility worker. It was determined that some of these postulated events (Events 28, 29, 31, 33, 330, 331, 332, and 335) have the potential for injury to the facility worker, and Tables 3.3-13 through 3.3-15 were annotated as low or moderate consequence (standard industrial hazard) according to Tables 3.3-8 and 3.3-9. These events included physical injury from falls, hand tools, steam, weld, and minor chemical burns.

The remaining E-4 events (Events 26, 26a, 27, 30, 32, 33a, 33b, 61, and 333), which do not involve radiological or toxicological hazards, were assumed to have the potential for causing prompt death to the facility worker. Therefore, Tables 3.3-13 and 3.3-14 were annotated as moderate consequence. These events included such industrial accidents as death from electrocution and drowning. Event 32 (falling into the basin) does pose a radiological contamination hazard, but any resultant dose from such a contamination would be small (negligible). The much more severe consequence from drowning (i.e., fatality) was used to derive the moderate consequence associated with this event.

Events 14 and 22

Several E-4 events have the potential for radiological impact to the facility worker. Events 14 and 22 involve the movement of fuel in the air, either from the cask or raising the fuel above the basin water level. It is assumed that the worst-case exposure, based on the RFA (worst-case), could result in a potential for a radiation exposure to the facility worker. It is assumed that this activity results in a low consequence to the facility worker as defined in Table 3.3-8, by the following mitigative feature. If these events were to cause an area radiation alarm to become activated, the facility workers would immediately evacuate the area, thereby mitigating the consequence.

Event 25

There are numerous occasions where fuel received is not as described on the shipping manifest (Ref. 35). Prior to shipping a cask containing irradiated reactor fuel elements to the RBOF, a complete description of the design and operation of the cask and its contents must be made available by the shipper for review by RBOF personnel. The RBOF personnel may then agree to accept shipment of the cask. However, it is possible for the RBOF to receive fuel that was not agreed upon. These errors between shipping manifest and fuel received can expose the RBOF facility worker to radiological hazards. If the shipping manifest does not agree with the preshipping descriptions, the cask cannot be unloaded without proper documentation and additional supervisory approval.

In the case that the shipping records do agree but during fuel unloading operations it is discovered the fuel identification does not agree with the cask packing list the fuel is immediately isolated in accordance with operating procedures. Once the fuel is isolated, corrective actions would be developed in accordance with approved procedures to determine the response to the unidentified fuel. Since the fuel cannot be unloaded until proper documentation exists or until the unidentified fuel is isolated, the impact of receiving improperly identified fuel to the facility worker is low consequence. Table 3.3-13 was annotated accordingly.

E-5: CRITICALITY

Due to large amounts of fissile material present in the form of irradiated fuel assemblies and due to the presence of water for moderation, criticality is possible in several areas of the RBOF basins. The postulated criticality can occur through process upset conditions (i.e., internal event initiators) or through NPH-induced conditions. The latter class of events includes seismic events, tornadoes, high winds, aircraft crash-induced initiators, and other external initiators outside the control of the facility.

Various fissile isotopes are contained in spent fuel stored in the RBOF (i.e., U-233, U-235, and Pu-239). Therefore, a study was conducted to determine which of these isotopes would yield the highest consequences during a criticality event. It was determined that the U-233 criticality

yields the highest inventory following a 60-second decay (Ref. 190) but that differences between U-233 and U-235 are negligible. For purposes of this document, the U-235 criticality is used as the basis for airborne doses to the facility worker and MOI. Direct dose to the facility worker is estimated using fission neutron and photon spectra from fission in U-235.

The consequences from the release of fission product inventory from criticality are assumed independent of the initiating event. In other words, once the fission yield is selected based on the historical record, criticality arguments and guidance in DOE-HDBK-3010-94 (Ref. 75), the resultant inventory of fission products is fixed when coupled with the inventory of the RFA. The process of estimation of downwind dose to receptors is therefore not sensitive to the type of initiating event.

In performing the criticality accident analysis, no credit is taken for the effect of burnup on reducing the possibility of criticality. In the actual case, it is clear that the fissile material content of the fuel is reduced and neutron poison content is present over that assumed condition of the RBOF fuel for the criticality studies. Additionally, fuel storage limits are set based on fresh, unirradiated fissile material content (i.e., burnup is not used to reduce margins for setting safe storage limits to preclude nuclear criticality).

An accidental nuclear criticality event has never occurred in the RBOF. Existing surveys of criticality events do not indicate any precedent for spent fuel storage facilities (Ref. 76). Nevertheless, criticality is postulated for purposes of evaluating the consequences from this accident condition to the facility worker, the individual at the OEP, and the MOI.

There are two major consequence concerns with the nuclear criticality event: (1) direct dose from prompt gamma and neutron radiation, and (2) dose from plume immersion (cloudshine and inhalation pathways). The total dose is the sum of the direct dose and the plume immersion dose evaluated at each receptor distance. Computed doses are based on an assumed level of fission in the criticality event.

Process-Induced Criticality (Events 24, 34, 35, 36, 37a, 37b, 38, 39, 40, 40a, 40b, 41, 42, 42a, 44, and 404)

Review of past criticalities due to process upset events and evaluation of internal events with frequencies above $1.0\text{E}-04/\text{yr}$ indicate that a yield of $5.0\text{E}+17$ fissions is bounding for the initial burst in an RBOF process-induced criticality (Ref. 77). Prototypic of these conditions are fuel storage infractions, cask overloading, bundling error, and mispositioning of fuel element baskets. In the fuel basket scenario, multiple elements may be spilled onto other fuel elements, but the criticality phenomenology again is bounded by a series of localized bursts each of magnitude $5.0\text{E}+17$ fissions.

As a bounding accident and facility receptor dose analysis, the hazards analysis assumed that the failed fuel releases all noble gas and halogen fission products from the $5.0\text{E}+17$ fission event into the basin water. Although some radiohalogens would be scrubbed in the process of transport to

the surface of the water, no credit was taken in the hazards analysis for this removal mechanism and all noble gas and radiohalogen fission products are released into the RBOF atmosphere. The facility worker thus receives dose from inhalation and shine due to the release of fission products over some period of time.

N-CLC-H-00112 (Ref. 78) details the analysis and the direct dose estimate for the facility worker calculated with the Monte Carlo N-Particle Transport Code (MCNP). Modeling of the postulated $5.0\text{E}+17$ fissions event assumes a 130-cm (4 feet 6 inches) overlying water layer and a total separation distance of 193 cm (6 feet 4 inch) to the facility receptor. Using conservative geometry assumptions and an assumed point-source criticality occurring at the top of the tallest assembly stored in the RBOF, the facility worker prompt dose from the initial burst was 12 rem. This assumed basin water level of 4 feet 6 inches was later included in Revision 0 of the TSR (Ref. 79).

Uncertainties associated with the modeling of hypothetical accident conditions at short distances within the facility are large and are heavily dependent on the assumptions made. In the hazards analysis, direct dose and the dose from fission gas inhalation and shine for 8 hours were summed to 30 rem as the overall radiological dose.

The postulated criticality event evaluation for the facility receptor numerically results in a moderate consequence level. This is the conclusion drawn comparing the 30-rem estimate with the criteria shown in Table 3.3-8.

Refer to Chapter 6 for the estimated dose to workers on the traveling bridge relative to the requirements defined in ANSI/ANS-8.3-1986 for a criticality alarm system.

Event 24

A criticality event was postulated in Table 3.3-13 in which credit cannot be taken for the attenuation factor of the water associated with the previous level of 4 feet 6 inches. Event 24 postulates a criticality occurring as a dry cask is being immersed in the basin. The event is estimated to be beyond extremely unlikely because of the rigorous procedural and design requirements of 10 CFR 71 (Ref. 31) for offsite fuel and because the onsite casks are received filled with water unless dry transfer is analyzed to be safe through the Unreviewed Safety Question [USQ] process). However, a criticality occurring during cask immersion is estimated to result in a substantially higher dose to the facility worker than calculated above (i.e., no credit can be taken for the shielding effect of the basin water). Therefore, Table 3.3-13 was annotated as moderate consequence.

Event 40b

The ability of RBOF operations personnel to verify the U-235 content of incoming foreign research reactor MTR fuel has been enhanced by of an assay device designed at Los Alamos National Laboratory. The Research Reactor Fuel Counter (RRFC) provides an assay chamber

and activation sources (1.2 Ci of Am-241 and lithium) to provide an indirect measurement of the U-235 contained in a measured assembly. The RRFC is normally stored in the cask basin and used in accordance with a sampling plan for incoming fuel shipments of MTR type fuel. To prevent the scenario in which an assembly is mispositioned under the RRFC while another assembly is being monitored by the RRFC, a mechanical barrier was installed at the bottom of the RRFC. (Ref. 228) Thus, the possibility of a criticality from such a scenario is beyond extremely unlikely, and Table 3.3-13 was annotated accordingly.

Event 37a and 36

The Systems Analysis (Ref. 34) suggests that the 100-ton crane or crane load can fall in the fuel storage basin due to mechanical failure. Specifically, the scenario (Event 37a) is developed as follows. The 100 ton crane is being used to move a load in the southward direction. The operator fails to follow established administrative controls and continues to move the crane toward the south travel limit switch. The south travel limit switch fails, allowing the crane to travel over the #2 storage basin, the crane continues to move until it reaches the end of the rail. The rail ends extend in to the vertical section of the building between the high and low bay areas. At the end of the rails, Drawing W233634 (Ref. 81) shows that there are passive, mechanical stops fastened to the crane rails to prevent the crane from reaching the rail ends. These stops were confirmed via walkdown. It is assumed that these stops prevent the crane from continuing past the end of the rails and falling into the main storage basin.

An additional concern is that once the crane contacts the mechanical stops, any load the crane may be transporting may fall into the #2 storage basin and deform the racks to the point that a critical configuration of fuel is obtained. This load is considered to be an offsite or onsite cask. To reduce the frequency of such an event, the limit switch is assumed to operate and prevent the crane movement beyond the #2 storage basin. It is therefore concluded that this scenario is extremely unlikely because of these mechanical barriers and the 100-ton crane south travel limit switch.

Event 36 is similar in that a process related criticality is postulated due to the crane and/or roof of the facility falling into the storage basin due to an external load dropped from a construction crane (see E-9 Extrinsic Events). The preventive features include the building design, the south travel limit switch preventing the 100 ton crane from being positioned over stored fuel, the fuel storage hardware and criticality control requirements which mandate initial safe storage of fuel arrays. It is therefore concluded that this scenario is extremely unlikely because of these mechanical barriers and the administrative controls of crane operations. Therefore, Table 3.3-13 was annotated as moderate consequence.

Event 404

This event postulates a criticality occurring as a result of an accumulation of fissile material on the basin purification filter or the deionizers, forming in an unsafe geometry. Due to the

following reasons, the accumulation of fissile material on the filter or in deionizers is not considered conducive to criticality:

- Basin chemistry samples show small amounts of alpha activity (Ref. 82). NMP-EHA-940036 (Ref. 82) showed no detectable fissile radionuclides in the resin bed. However, the resin beds do accumulate other radionuclides which have significant thermal neutron cross-sections and emit gamma radiation. Since the deionizer resin curie loading is monitored and the resin is periodically replaced (Ref. 58 and 83), it is very unlikely that the minimum theoretical amounts of fissile radionuclides necessary for criticality could accumulate before other radionuclides caused the deionizer resin to be replaced.
- Assuming a gross fuel failure (such that massive amounts of fissile material are released and deposited on the deionizer), a criticality event would require that the particulates are captured in the deionizer or filter suction piping. This piping is suspended from the walls of the basin and is not capable of intaking any significant particulate accumulation which would sink to the basin floor.

Therefore, a criticality event in the purification filters is estimated to be beyond extremely unlikely. Table 3.3-15 was annotated accordingly.

Event 42a

A criticality event was postulated in Table 3.3-13 in which credit cannot be taken for the attenuation factor of the water associated with the previous level of 4 feet 6 inches. Event 42a postulated a criticality occurring as a result of a fast water leak in the 70-ton shipping cask. Some types of fuel may experience a positive reactivity transit during a rapid draindown event. A criticality occurring in a shipping container is estimated to result in a significantly higher dose to the facility worker than calculated for a criticality occurring in the basin. Therefore, Table 3.3-13 was annotated as moderate consequence.

NPH-Induced Criticality (Event 43)

This scenario postulates leakage of the basin water to the subsoil due to an earthquake resulting in rack failure and fuel deformation. This deformation results in fuel melting and a criticality with fission product release without basin water decontamination. It is assumed that the source term will be due to criticality and fuel melting with cladding breach. The overall consequences then are the sum of those due to criticality and those from the melting of the fuel.

To determine the consequence, the hazards analysis assumed that the NPH-induced event causes the fuel cladding to breach, releasing volatile and non-volatile fission products to the atmosphere.

This potential release results in high consequences at the site boundary; therefore, Table 3.3-13 was annotated accordingly.

E-6: INTERFACES

During normal RBOF/RRF operations, numerous interfaces take place between systems, equipment, personnel, and various combinations of the three. The RBOF and RRF contain hazardous energy sources and hazardous material sources capable of causing failures between these interfaces during normal routine operations. Therefore, events involving these hazards were postulated for each of the five evaluation areas. Reviewing Tables 3.3-13 through 3.3-17, it can be seen that these E-6 events include such events as material incompatibility, loss of services, and fire.

Events 45, 46, 47, 48, 49, 337, 338, and 339

For postulated E-6 events considered industrial in nature involving loss of services (Events 45-49 and 337-339), where radiological or toxicological hazards are negligible, the consequences were based on the potential for physical injury to the facility worker. It was conservatively assumed that these postulated industrial type events have the potential for minor injury to facility workers. Tables 3.3-13 and 3.3-14 were annotated as low to negligible consequence.

Events 15, 54, 55, and 319

Events 15 (loss of nitrogen during cask cooling), 54 and 55 (truck/rail unable to ship fuel out of the RBOF), and 319 (loss of instrument air to chemical pad) have no health and safety consequences. Tables 3.3-13 and 3.3-14 were annotated as negligible consequence.

Event 51

The interaction of basin water and offsite fuels containing sodium or other materials which react with water could result if the primary containment of fuel canisters is compromised. To determine the consequence, it is assumed that this postulated event causes the fuel cladding or primary containment to leak or breach, releasing fission products to the basin water. It is further assumed that 30% of the noble gas Kr-85 inventory from the damaged fuel assemblies (RFAs) is released to the atmosphere. Other noble gases will not be present in significant quantities due to decay, or do not contribute appreciably to the radiological dose. The radiological consequences can be bounded by the case where all 2,800 fuel assemblies are assumed to be damaged. The resultant dose to the individual at the OEP using average meteorology is less than 0.5 rem (see Event 36). The dose to the facility worker would be expected to be significantly higher but still in the low consequence category (<25 rem due to radiation monitoring) as defined in Table 3.3-8. However, life-threatening injuries from the violent reaction or secondary effects could occur to

the facility worker, and the event is classified as a moderate consequence event as defined by Table 3.3-8. Table 3.3-13 was annotated accordingly.

Event 53

A malfunction of an area radiation alarm, in itself, offers no consequence to the facility worker. Fuel handling activities are not allowed in the RBOF without appropriate radiation monitoring protection. Therefore, Table 3.3-13 was annotated as no consequence.

Events 56 and 340

The postulated fires beginning outside of the RBOF/RRF area and spreading to the RBOF and RRF (Events 56 and 340) are considered to have no to low impact on the health and safety of the facility workers. If a major fire was in a neighboring area, there would be time to evacuate facility workers before exposing them to any significant health or safety risks. The applicable safety documentation (facility in which the fire originated) covers the consequence to that facility/site worker and the public. Therefore, Tables 3.3-13 and 3.3-14 were annotated as no to low consequence to the RBOF/RRF worker.

E-7: WASTE DISPOSAL

During normal operations in the RBOF and RRF, both hazardous and nonhazardous waste is produced in the form of solid and liquid wastes as identified in Table 3.3-6..

Events 58 and 342

Previous facility analysis (Ref. 34) also documents the potential for improper disposal of solid waste (Events 58 and 342). These events involve the packaging of contaminated waste by RBOF/RRF personnel into a clean dumpster, instead of a regulated dumpster, and subsequent transfer to the sanitary landfill. These events do not have the potential for physical injury to the RBOF/RRF worker. However, the potential exists for possible contamination of landfill workers. Therefore, evaluating a consequence level for these events is outside the scope of this SAR. On this basis, Tables 3.3-13 and 3.3-14 were annotated as no consequence as it pertains to the RBOF and RRF.

Events 57 and 341

Event 57 involves the inability to transfer waste from the cask wash pit to the waste tank. This event has a negligible health and safety consequence because the transfer takes place from the cask wash tank to the shielded waste cell through pipe embedded in concrete and because the

inability to transfer does not result in a loss of containment. Therefore, the health and safety consequence is negligible, and Table 3.3-14 was annotated as accordingly.

Event 341 involves the inability to transfer waste from the RRF to the waste tank. This event has a negligible health and safety consequence because the transfer takes place from the shielded RRF cell to the shielded waste tank and because the inability to transfer does not result in a loss of containment. Therefore, the health and safety consequence is negligible, and Tables 3.3-13 and 3.3-14 were annotated as accordingly.

E-8: MAINTENANCE

As noted in Tables 3.3-4 and 3.3-5, the RBOF and RRF contain hazardous energy sources and hazardous material sources capable of causing injury to the workers during normal maintenance activities. Therefore, maintenance events involving these hazards were postulated for each of the five evaluation areas.

Events 59 and 59a

During routine maintenance operations performed in the RBOF and RRF, it is possible for a facility worker to be exposed to varying levels of radiation or contamination (Events 59 and 59a). Maintenance activities are carefully planned using As Low As Reasonably Achievable (ALARA) guidelines. As noted in previous facility analysis (Ref. 34), the worst cases of radiation exposure and contamination have been from removing equipment from basins and working on pumps in the basin. It is assumed that all maintenance activities that result in radiation or contamination would be controlled by the Radiological Protection Program, and Table 3.3-13 was annotated as low consequence to the facility worker.

Event 60

As noted in the 200-Area Data Bank (Ref. 84), small fires have occurred when welding activities have ignited plastic covers. These fires were usually self-extinguishing due to the extremely low fire loading or were quickly extinguished through hand-held fire extinguishers. Therefore, Table 3.3-13 was annotated as low consequence (standard industrial hazard).

Event 343

During routine maintenance activities on systems that contain chemicals, the potential exists for the facility worker to be exposed to varying amounts of hazardous chemicals. Scenarios involving chemical releases have been previously analyzed and are discussed elsewhere in this chapter. The worst-case hazardous chemical scenarios (Events 316, 318, and 328c) are assumed to be the worst-case scenario for maintenance activities involving hazardous chemical releases,

and discussions are be repeated here. Therefore, Table 3.3-14 was annotated as moderate consequence.

Event 52

A NIM alarm malfunction, in itself, offers no consequence to the facility worker. Fuel handling activities are not allowed in the RBOF without appropriate radiation monitoring protection. Therefore, Table 3.3-13 was annotated as no consequence.

E-9: EXTRINSIC HAZARDS

As noted in Table 3.3-6, the operation of RBOF and RRF involves hazardous materials. While this material (excluding the chemical hazardous material) is maintained in primary containments and confined by the building structure and ventilation system, extrinsic hazards have the potential for the following:

- Damaging the primary containments
- Affecting confinement by damaging the RBOF and RRF structurally
- Turning the critically safe configuration of stored fuel into critically unsafe configurations
- Releasing chemical inventories

Therefore, events involving these hazards were postulated for each of the evaluation areas. Three categories of extrinsic events germane to the operations were identified: aircraft crash, ground vehicle accident and external impacts from loads dropped from construction cranes. Adjacent fires and explosions were eliminated from consideration because nearby facilities are far enough away so as not to impact this area (Ref. 48).

Events 62 and 345

Tables 3.3-13 and 3.3-14 estimate the frequency of aircraft crashes (Events 62 and 345) to be incredible ($<1E-06$ occurrences/yr). Because of this estimate, a reasonable scenario is not conceivable (this frequency analysis is conservative and is not subject to the $1E-07$ /yr cutoff). Therefore, it was concluded that aircraft crashes are beyond extremely unlikely, and the corresponding tables were annotated accordingly.

Events 345a, 345c, and 345d

A ground vehicle (e.g., truck, car, forklift) crashing into a trailer/portable deionizer staged at hose box No. 1 during resin regeneration (Event 345a); chemical pad No. 1 (Event 345c); or chemical pad No. 2 (Event 345d) has the potential for damaging containments. Based on the physical

layout of the area and facility, and posted speed limits, reasonable worst-case scenarios would be that releases comparable to those for loss of containment events previously discussed (Events 324, 316, and 318, respectively) would occur. On this basis, Table 3.3-14 was annotated accordingly.

Event 36

A deformation of the fuel storage racks resulting in a criticality was postulated from the disruption of the facility roofing material and/or the 100-Ton crane due to a load dropped from a construction crane. This is further discussed in Section E-5 Criticality. Due to the potential criticality, this event was assigned moderate consequence and Table 3.3-13 was annotated accordingly.

E-10: NATURAL PHENOMENA

As noted in Table 3.3-6, the operation of RBOF and RRF involves hazardous materials. While this material is maintained in primary containments and confined by the building structure and ventilation system, it has been shown in previous sections that natural phenomena have the potential for the following:

- Damaging the primary containments
- Affecting confinement by damaging the RBOF and RRF structurally
- Turning the critically safe configuration of stored fuel into a geometry with the potential for criticality
- Rupturing chemical tanks and thus exposing their inventory to the environment

Therefore, events involving these hazards were postulated for each of the evaluation areas established. These E-10 events, in most cases, are repeats and summaries of events discussed in previous sections. Therefore, where possible, the health and safety consequences are based on prior discussions.

Events 63 and 64

During development of the hazards analysis, the structural analysis of the RBOF and RRF was ongoing. For conservatism and on the basis of previous analysis (Ref. 46, 85), the building structure was assumed to fail for any specific seismic value in the hazards analysis. (For analysis relative to the completed structural assessment of the RBOF and RRF, refer to Section 3.4.)

The criticality event category (E-5) discusses the consequences of criticality events caused by natural phenomena. Based on that discussion, Table 3.3-13 was annotated as moderate consequence.

Event 63a

This event postulates an NPH event causing a criticality while the basin is in a draindown condition. This combination of events may result in fuel melting during and/or after the criticality event and has significant offsite consequences. Therefore, Table 3.3-13 was annotated as high consequence.

Event 347

It is anticipated that the RRF or the chemical pads could receive some damage during a PC-2 0.15g earthquake or during 84-mph high winds, resulting in a loss of chemicals or contaminated water as previously discussed (Ref. 85). It is anticipated that if an earthquake or high wind event occurred, the facilities and area would be evacuated immediately and re-entry would be performed as required by procedure. However, the potential exists to exceed moderate consequence level threshold values, and Table 3.3-13 was annotated accordingly.

Event 347a

It is anticipated that the RRF or hose rack could receive some damage during a PC-2 0.15g earthquake (Ref. 25) or during 84-mph high winds, resulting in a loss of radionuclide contaminated water and chemicals. It is anticipated that if an earthquake or high winds occurred, the facilities and area would be evacuated immediately and re-entry would be performed as required by procedure. It was estimated that this event would release less than the low consequence threshold value of radionuclides and would result in a low consequence from any chemicals released. Table 3.3-13 was annotated accordingly.

Event 65

A recent hazards evaluation concluded that the only missile of concern for a PC-3 facility is assumed to be a 3-inch-diameter steel pipe with a material density greater than water, with a 35-mph vertical missile velocity (Ref. 85). NSDSs allow the spacing between the fuel elements to be less than 1 foot, which makes it probable that more than one fuel element would be breached from a potential missile. This event is bounded by the breach of all 2,800 fuel assemblies releasing 30% of the Kr-85 discussed previously (Events 7, 10, and 11).

Evaluation of tornado-induced criticality at the RBOF is confined to accident scenarios that are initiated by wind-induced structural failure. A separate missile hazard evaluation is not included. Direct missile penetration of the below-grade storage basins is not considered credible, and the occurrence of debris impingement resulting from missile-induced structural collapse is bounded by direct wind damage to the structure. It is not considered credible that a missile impact can cause sufficient disruption of storage rack geometry to cause a criticality. Therefore, it was

concluded that this event results in moderate consequence to the facility worker as previously discussed, and Table 3.3-13 was annotated accordingly.

Events 9 and 66

Event 9 (previously discussed in the Loss of Containment section) determined the consequences of basin overflow following a loss of containment event. Event 66 postulates a basin overflow due to heavy rains and a leaking roof. While initiated in a different manner, this event yields the same consequences as Event 9. Therefore, based on the basin overflow, it was concluded that Event 66 resulted in negligible consequences to the facility worker as defined in Table 3.3-8, and Table 3.3-13 was annotated accordingly.

Event 67

Large accumulations of snow and ice (which could potentially collapse the roofs) are quite rare due to the warm climate at SRS. Table 2.3-2 of WSRC-SA-10003 (Ref. 223) lists the recurrence intervals of ice accumulation at SRS. From this table, the amount of ice present at a recurrence interval of 1,000,000 years (i.e., frequency of $1\text{E-}06/\text{yr}$) is less than the Building 244-H roof can support. Therefore, this event can be considered beyond extremely unlikely, and Table 3.3-13 was annotated accordingly.

Events 68 and 346

The principal adverse effect of a meteorite crash (Events 68 and 346) is loss of containment, which has negligible and moderate consequences, respectively, to the facility worker. However, these events are beyond extremely unlikely, and Tables 3.3-13 and 3.3-14 were annotated accordingly.

Events 69 and 349

The principal adverse effect of lightning (Events 69 and 349) is interruption of electrical power and has negligible health and safety consequence to the facility worker. Tables 3.3-13 and 3.3-14 were annotated accordingly.

Event 348

Snow, ice, or hail (Event 348) may adversely affect operations on the chemical pads. Currently, snow and ice and accompanying low temperatures may burst valves or pipes, instrumentation may become ineffective, and equipment with movable parts may become immobilized. The worst-case effect from snow, ice, or hail is postulated to be a pipe, hose, or valve break releasing minor quantities of chemicals onto the chemical pad. Scenarios involving minor chemical

releases have been discussed. Therefore, based on the previous discussion in the Loss of Containment section, Table 3.3-13 was annotated as moderate consequence.

Event 401a

Nitric acid is stored in tank EP50, which is located on chemical pad No. 1. Sodium nitrite is stored in two tanks located on chemical pad No. 2. In the event of an earthquake or wind-induced failure of the rain shelter, these tanks or associated piping could leak or breach and release their contents onto their respective chemical pads. Chemical pad No. 1 gravity-drains to waste tank EP38. Chemical pad No. 2 sump is pumped to waste tank EP38. Given the extremely unlikely chance that the pump and controls survive the earthquake or rain shelter failure, the leaking sodium nitrite could be pumped to waste tank EP38 and mix with the nitric acid. According to S-CLC-H-00203 (Ref. 25), the mixing of sodium nitrite and nitric acid is reactive and can generate a large volume of brown nitrous dioxide (NO_2) fumes. The temperature would also increase in waste tank EP38 (Ref. 25). Nitrous dioxide ERPG values are relatively low, 15 and 30 ppm for ERPG-2 and -3, respectively.

Therefore, to reduce the risk associated with this scenario, a PMT (Ref. 38) was developed to modify the sump pump located on chemical pad No. 2. This modification installed a disconnect switch on the pump motor power circuit and allowed de-energization of the pump motor under normal operating conditions. The PMT also modified the pump discharge piping to provide a means of isolating the pump discharge during normal conditions. Under unlikely conditions where the sodium nitrite tank leaks, motor power and pump discharge can be provided and the leak can be pumped to waste tank EP38. This will require verification of the contents of waste tank EP38 prior to pumping the spilled chemical.

Calculations have been performed to determine the maximum volume that the berm around chemical pad No. 2 can contain (Ref. 39). This volume is 963 gallons and will require the chemical inventory to be limited to 900 gallons as specified in Table 3.3-4. The berm around chemical pad No. 2 has also been qualified to withstand both the collapse of the rain shelter without cracking and PC-2 seismic loads (Ref. 40, 73). The nitric acid and sodium nitrite hold tanks and sodium nitrite dumpster have been qualified to all PC-2 seismic and wind loads (Ref. 41). Given the above administrative and design controls, the frequency of such an event is beyond extremely unlikely (Ref. 42). Table 3.3-15 identified this event as Event 401a.

PLANNED DESIGN AND OPERATIONAL SAFETY IMPROVEMENTS

The following discussion represents the modifications to the RBOF and RRF to improve or upgrade the supporting systems.

As part of the structural analysis of the fuel racks for the main and #2 storage basin, two modifications were implemented to increase the capacity of the racks to the loads associated with

a PC-3 seismic event. These modifications are associated with the bundle storage racks and the PM-3A hardware currently used on Row UU in the main storage racks (Ref. 86, 87).

Modifications have been implemented (Ref. 38) to modify the sump pump located on chemical pad No. 2. This modification installed a disconnect switch on the pump motor power circuit and allowed de-energization of the pump motor under normal operating conditions. The PMT also modified the pump discharge piping to provide a means of isolating the pump discharge during normal conditions.

A conceptual design change has been issued to upgrade the current breathing air system. This upgrade will replace the single 25-hp breathing air compressor with two 60-hp air compressors and associated equipment (Ref. 88). A new building will be constructed to house the two existing instrument air compressors, the existing air receiver, and the two new breathing air compressors.

Design Change Package E-DCP-H-96072 (Ref. 224) has been issued to provide enhanced safety modifications for the 100 ton crane. The modifications will provide an emergency stop pushbutton and defeat the fast travel mode in the south direction for the bridge crane.

A removable racktie has been proposed (Ref. 225) to provide the north-south restraint between the RBOF fuel rack frames when normal engagement of the permanent rackties is not possible. This misalignment may occur due to the flexibility of the RBOF fuel storage racks during loading/unloading and repositioning of fuel.

A proposed scope-of-work summary was developed to do the following:

- Identify the system components whose failure could result in a facility shutdown of an unacceptable time interval
- Identify the upgrades that have the greatest potential impact on the safety of the facility
- Identify upgrades that could bring the RBOF facility into compliance with DOE Orders, national consensus codes, and standards
- Identify upgrades that could promote facility independence and greatly impact the operability of the RBOF facility

As documented in EFR-RSE-96-0051 (Ref. 89), several upgrades have been proposed, including improvements in waste tank EP38, the filter-deionizer system, RRF, ventilation system, standby power, and 100-ton crane.

DEFENSE IN DEPTH

Defense in depth as an approach to facility safety builds in layers of defense against the release of hazardous and radiological material so that no one layer is completely relied upon. To

compensate for potential human and mechanical failures, defense in depth is based on several layers of protection with successive barriers to prevent the release of hazardous materials to the environment. This approach includes measures to protect the public, site worker, and the environment from harm in case these barriers are not fully effective. Defining defense in depth as it exists at a given facility is crucial for determining a safety basis. This section identifies the broad purpose of defense in depth used to support the conclusion that defense in depth for a given hazard is commensurate with the industrial practices for the type of specific activities taking place within the facility.

Typical categories for lines of defense include the following:

- Physical barriers to the release. This category may include barriers to contain uncontrolled hazardous releases (e.g., process piping and storage tanks) or energy releases, including barriers which would only come into play if other barriers were to fail. There is no limit to the number of physical barriers that could provide protection for a given hazard. Each barrier can be considered a separate line of defense as long as it is properly independent of, and in series with, the other lines of defense for the hazard.
- Engineered systems that can function to limit the value of some operational parameter (e.g., the ventilation system can reduce the concentration of explosive vapors) or to mitigate uncontrolled releases of hazardous material or hazardous energy upon barrier failure. The engineered systems do not have to be specifically designed to provide a safety function; however, they must be reasonably expected to provide whatever functionality is claimed.
- Administrative controls, including procedural restriction and limits, requirements for manual monitoring of critical parameters, equipment support actions, and responses or actions credited to limit abnormal conditions, accident progression, or potential personnel exposure.
- Emergency procedures, including civil defense procedures

All assumptions (including those related to inventory control and initial conditions) are evaluated as potential lines of defense in the categories listed above.

The methodology for identifying those events requiring defense-in-depth features for the RBOF involves a hazard screening process. The hazard screening process is similar to that used in the hazard evaluation provided in Section 3.3.2.3. However, the identification of defense-in-depth features also involves an evaluation of the potential for adverse effects on the facility worker. This evaluation is provided in the RBOF hazards analysis (Ref. 25).

The purpose of the hazard screening process is to identify the potential events that result in situations of major concern or situations of concern. Thus, potential events that result in situations of only marginal concern are eliminated from further consideration, and the identification of the defense-in-depth features is not required. Therefore, based on the hazard

evaluation, potential events that result in high consequences and that are considered credible (i.e., the frequency is greater than $1.0\text{E-}06$ events/yr) and potential events that result in moderate consequences and that are considered unlikely (i.e., the frequency is greater than $1.0\text{E-}04$ events/yr) are evaluated and defense-in-depth features are identified. However, as discussed in Section 3.3.1.1, standard industrial hazards are also eliminated from further consideration, and the identification of defense-in-depth features is not required. An exception to this methodology involves the evaluation of defense-in-depth features for nuclear criticality events. All nuclear criticality events are evaluated for defense-in-depth features regardless of the frequency.

The screening process performed evaluated the 136 postulated events identified for the RBOF and RRF. The process is documented for each event on a hazard screening form. Each event was screened as either positive or negative. Moderate or high consequence events with a probability of occurrence greater than $1.0\text{E-}06$ events/yr are screened as positive. Events that were identified as containing inherent assumptions were also screened as positive events. All criticality events are screened as positive. The screening process evaluated all 136 postulated events and screened 81 as positive events. Of these, 37 positive screenings were based on consequence binning, and two were based on criticality regardless of consequences or frequency. The remaining 44 events were positively screened based on inherent assumptions. The remaining events were determined to represent only low or negligible hazards and do not require additional quantitative accident analysis.

Defense-in-Depth Design Features

The events that result in a major concern for the worker at the RBOF were evaluated to identify defense-in-depth design features associated with the prevention and mitigation of these events. These accidents include fires, explosions, spills, radiological and hazardous material exposure, and criticality involving various initiators. An examination of these events results in the identification of several features that provide defense in depth for each of the events. These features are characterized as barriers, preventive and mitigative systems, instrumentation and alarms, and administrative controls and were derived from the hazards analysis and Chapter 2. Chapter 2 provides further detail of the design features for the RBOF facility. The following is a list of available defense-in-depth features for the RBOF and RRF (refer to Chapter 4 for the explicitly credited SSCs):

Barriers, Preventive Features, and Mitigative Features

- RBOF/RRF building structure
- Basin structure (stainless steel lined floor and phenolic lined walls) and canal system
- Shielding
- Basin water (makeup system)
- Offsite and onsite casks

- Cask coolant
- Cask inserts (to maintain fuel spacing during transport), spacers, grates, RHF inserts, HFIR inserts, and blanks
- Fuel cladding
- Remoteness of the facility (relative to the site boundary)
- Storage tanks
- Berms and dikes around chemical pads
- Filter-deionizer system (tanks EP14, EP15, and EP48, and embedded piping)
- Resin regeneration tanks and embedded piping
- Waste tank EP38, agitator, embedded piping, and waste sampler
- Crane lifting/moving devices and fixtures:
 - Bridge crane
 - Traveling bridge
 - Monorail system
 - Mechanical stops
 - Crane south travel limit switch
- Operating equipment design (saw table stops, measurement rods)
- Storage racks
- Ventilation system (exhaust and vessel vent system) (basin areas at lower pressure than rest of building):
 - Building exhaust filters
 - Stack
 - Ventilation ducting
- SRS security fences and gate
- Railing, gratings, and partitions around and over basin
- Backup water supply
- Emergency generators
- Sodium fuels stored in air-tight containers
- Acid metering pump
- Vessel venting
- Drains to waste tank
- Miscellaneous hardware:

- Buckets (bucket inserts)
- Bundles
- Lugs
- Guide plates
- Grates
- Spacers (swingdown and removable)
- Rack ties
- Row latches
- I-beams
- Manual fire suppression
- Life preservers

Instrumentation and Alarms

- Basin water level instrumentation
- Basin temperature instrumentation
- Tank level instrumentation
- Kanne monitor
- Gamma monitors
- Continuous air monitors
- Personnel contamination monitors
- NIMs
- Basin water level alarms
- Tank level indicator
- Anticontamination clothing
- Plastic suits
- Emergency breathing apparatus

Defense-In-Depth Administrative Features

- Source term controls
- Basin water activity limits
- Storage tank inventory limits

- Hazardous Material Inventory Control Program
- Fire Protection Program
- Radiological Protection Program
- Fuel Receipt and Shipping Program:
 - Fuel physical characteristics and integrity
 - Maximum radionuclide inventory
 - Sodium content
 - Maximum cooling time
 - Maximum heat load
 - Deflagration concerns
 - Storage location
- Industrial Safety and Hygiene Program
- Emergency Response Program
- Installed Process Instrumentation Program
- Configuration Control Program
- Nuclear Criticality Safety Program
- Quality Assurance Program
- RBOF Deionizer Resin Replacement/Regeneration Program (Ref. 83)
- 10 CFR 71 packaging and transportation of radioactive material
- 49 CFR 173 IAEA Safety Series No. 80
- Worker Training and Qualification Program
- Criticality/fuel handling procedures:
 - Mass controls
 - Spacing controls
- Operating/emergency procedures
- Fire response procedures
- Basin Water Chemistry Control Program
- Combustible loading
- Explosive restrictions
- Freeze Protection Program
- MSDS

- NSDS
- Structural Integrity Program

Defense in Depth - Process-Induced Criticality

To demonstrate the defense-in-depth concept of providing multiple and/or parallel sets of systems and/or mitigating the potential for releases of hazardous energy and hazardous material, the following discussion provides a logical overview of the various lines of defense for process-induced criticality.

Part of the RBOF mission is to receive and store a wide variety of nuclear fuel assemblies. Handling includes removal of assemblies from shipping containers, placement of fuel assemblies into shipping containers, placement in the storage pool, and movement around the storage pool. The fuel assemblies are moved by means of a traveling bridge and monorail system and are stored in lockable storage racks. This accident scenario involves a criticality event due to fuel handling or storage infractions. Due to the nature of operations performed in the RBOF, there are many events that have the potential to lead to a criticality. Further details regarding the potential dose to facility workers from a process-induced criticality are provided. The following are major event initiators and lines of defense for a criticality event.

Human errors during the following activities could lead to a criticality:

- Placement of fuel
- Loading/unloading of fuel
- Relocation of fuel
- Movement of fuel between storage and working basins
- Loading of transfer buckets during cask unloading operations

In addition, the following initiators could lead to a criticality:

- Dropping of fuel bundles, assemblies, or transfer buckets
- Movement of fuel containers to within 12 inches of one another
- Overloading of a shipping cask
- Floating fuel containers
- Breached fuel containers

Because the criticality evaluates a range of initiating events, the following lines of defense are recommended from a broad aspect and may not address each specific initiator:

- Storage racks ensuring spacing and configuration (preventive)

- NSDSs to provide the basis for fuel spacing and mass requirements (preventive)
- Worker training (specifically in the safe operation of cranes and safe handling of fuel assemblies (preventive)
- Fuel cladding
- Fuel Handling Equipment
- Reactor burnup
- Fuel storage in a water-filled basin under a minimum water level
- Fuel Receipt and Shipping Program
- Building exhaust filters
- Area radiation monitoring systems (NIMs, GA-6 monitors, stack monitors)
- Emergency response plans
- Worker training (specifically in safe response to a criticality)
- Double contingency analysis

Safety-Significant Structures, Systems, and Components

SSCs that are major contributors to defense in depth or worker safety, as determined in the hazards and accident analysis, are designated as safety-significant SSCs through the functional classification process. The RBOF features that are designated as safety-significant SSCs are as follows:

- Storage racks and associated hardware
- Fuel basin and canal system
- Basin water level indicator (ruler)
- Shielding
- Onsite casks
- Area radiation monitoring system (NIMs)
- Berm at chemical pad No. 2
- Crane limit switch system
- Building 244-H structure
- Nitric acid tank
- Sodium nitrite tank
- Sodium nitrite hold tank
- Remote alarms

- RRFC skirt
- Fuel transport and storage equipment

Chapter 4 discusses these safety-significant SSCs in detail.

Technical Safety Requirements

The defense-in-depth features that require TSR coverage are those SSCs listed previously, as well as the following safety-class administrative controls:

- Fuel Receipt and Shipping Program
- Hazardous Material Inventory Control Program

The following are safety-significant administrative controls:

- Nuclear Criticality Safety Program
- Waste Management Program
- Hoisting and Rigging Program
- Radiological Protection Program
- Emergency Preparedness Program
- Basin Water Chemistry Control Program
- Deionizer Regeneration/Replacement Program
- Fire Protection Program
- Worker Training and Qualification Program
- Configuration Control Program

WORKER SAFETY

Design and administrative features that protect workers in the RBOF are listed as “preventive features” and “mitigative features” in Tables 3.3-13 through 3.3-17. Many of the features protecting workers contribute to the defense in depth of the facility and are discussed in the preceding section. In addition, many of these features are considered as candidates for designation as safety-significant SSCs as well as for TSR coverage. Features identified as safety-significant SSCs and/or protected through TSRs are identified in Chapter 4.

This section summarizes the major features protecting workers from the hazards of facility operation, exclusive of standard industrial hazards. The major features protecting workers from the hazards of facility operation at the RBOF and RRF include the following:

- Shielding
- Basins and associated transfer canals
- NIMs
- Chemical pad No. 2 berm
- Storage racks and associated hardware
- 100-ton crane south travel limit switch
- Nitric acid and sodium nitrite storage tanks
- RRFC skirt
- 244-H building (includes crane south travel mechanical stops)
- 70-ton onsite shipping cask
- Fuel transport and storage equipment
- Remote alarms

These features are designated as safety-significant SSCs for worker safety because their failure to perform their safety-significant function is estimated to result in life-threatening injuries to workers. Serious injuries refer to immediately life-threatening or permanently disabling injuries (e.g., loss of eye or loss of limb) from a radiological accident.

Safety-significant SSCs for worker safety are a part of a much larger set of safety features including administrative features. This larger set of worker safety features includes the following, which are not safety-significant SSCs:

- Portable fire extinguishers
- National Fire Protection Association-approved storage cabinets for chemicals
- Retaining chains and caps for gas cylinders
- Emergency lighting
- Portable respiratory devices
- Breathing air system
- Public address system
- Safety showers
- Eyewash stations
- Protective clothing, including gloves
- Approved scaffolding, rigging, and associated safety equipment
- Tool and equipment guards
- Hearing protection

- Equipment grounding devices
- Ground fault interrupters

The safety-significant SSCs and nonsafety-significant SSCs identified above generally protect the facility workers who are involved with RBOF/RRF work activities. The major types of activities performed at the RBOF and RRF include the following:

- Cask handling (i.e., receipt and shipment of fuel)
- Movement of fuel (i.e., storage, transfer, inventory, disassembly, and repackaging)
- Fuel assembly cropping
- Resin addition, regeneration, and spent resin transfer
- Chemical and resin sampling of basin water
- Decontamination and housekeeping
- Chemical handling and mixing operations
- Waste handling and processing

Worker Criticality Safety

Depending upon the operations being performed, additional protective measures are necessary to ensure adequate safety for the workers who can be directly exposed to hazards. Hazards involving radiation exposure and criticality greatly impact worker safety. Workers at the RBOF and RRF are trained to respond to area radiation alarms and to follow appropriate evacuation procedures. These actions minimize the radiation dose received. During fuel handling operations, workers performing these activities are trained to follow procedures. Administrative programs and precautions taken to prevent and mitigate radiological hazards include the following:

- Radiological Protection Program
- Radiation and contamination training
- Radiation monitoring
- Fuel handling and storage procedures
- Fuel Receipt Program (including radiation survey and receipt inspection)
- Use of 10 CFR 71 qualified offsite casks (Ref. 31)
- Fuel cropping procedures
- Basin water level control
- Water chemistry control limits
- Cask cooling procedures

- Criticality training for RBOF personnel
- Criticality Control Program
- Control of basin decay heat load
- Stripping of sodium from fuel before shipment to SRS
- Emergency and disaster plan training

The above controls and administrative programs are the principal features used to prevent a criticality event from occurring. Further details associated with controls are provided in the double contingency discussion of Chapter 6.

Given the unlikely occurrence that a process-induced criticality did occur in RBOF, the dose to the facility worker was evaluated (Ref. 90). However, in such an analysis the fission yield, initial conditions, and status of both active and passive features in the facility may vary substantially for a criticality event in a spent fuel storage facility, thus contributing to large uncertainties associated with the source term and consequences of a process-induced criticality. Recognizing these factors, the various scenarios discussed below attempt to bound the potential impacts.

Given the unlikely occurrence of a process-induced criticality, the estimated dose to an individual in the facility will be composed of direct radiation from the criticality burst, "cloudshine" (direct radiation from the released fission products in air), and inhalation components. While the direct radiation and shine components are largely due to fission gases generated from the criticality event, the inhalation component can be dominated by low volatility radionuclides. These radionuclides, such as strontium, ruthenium, and plutonium, will not be released in appreciable quantities unless significant melting of one or more fuel elements occurs. In this analysis, and in the analysis supporting Section 3.4 as well, it is conservatively assumed that melting of the fuel cladding follows the criticality burst. The melting releases residual, long-lived activity present in the fuel. Much of the long-lived activity is dose-dominant and includes Ru-106 and Sr-90, as well as several transuranic isotopes. Thus, release of dose-dominant isotopes is included in the overall inhalation component.

Melting of fuel material following a criticality is dependent on the reactivity insertion rate, fuel geometry, heat transfer conditions, and fuel type. Of the 41 accidents included in the review of criticality events by Stratton/Smith (Ref. 76), there is mention or evidence that melting of the fuel ensued in seven cases. However, of the seven cases, two were less than $1.0\text{E}+18$ fissions. Both of these criticalities were in systems unrepresentative of those contained in the RBOF: the LLNL event in 1963 was bare reflected metal ($3.7\text{E}+17$ fissions), and the EBR-I event in 1955 was a fast reactor with NaK ($4\text{E}+17$ fissions).

Williamson (Ref. 77) has identified additional data from the SPERT tests indicating melting occurred, associated with fission yields below $1.0\text{E}+18$ fissions. The amount of melting varied as a function of the estimated fission yield: (1) 0.5% of the core ($3.5\text{E}+15$ fissions), (2) ~3% of the core ($6.1\text{E}+17$ fissions), and (3) ~35% of the core ($9.9\text{E}+17$ fissions). Although these tests were somewhat more representative of RBOF conditions, the SPERT tests involved fuel

constrained in a critical geometry and deliberate insertion of large reactivity. Under the process-induced scenarios considered for the RBOF, the SPERT conditions are not likely.

It is recommended that if the criticality level for the RBOF is chosen below $1.0E+18$ fissions, then logic models for the RBOF for a process-induced criticality are justified in applying a 0.1 conditional probability for melting, since there is a preponderance of experimental evidence supporting a "no-melt" outcome (Ref. 91). The existing fault trees for determining an overall process-induced criticality frequency (Ref. 188) do not explicitly model the potential accident progression once the event occurs. For conservatism, the accident analysis in Section 3.4 assumes that if the criticality event occurs, the affected fuel will melt with no dependence on the event initiator. For the bounding analysis in Section 3.4, the fuel melting will be assumed to occur at the same frequency as the criticality event itself. However, for assessing the potential impact to the facility worker, process-induced criticality frequencies specified in Section 3.4 will be reduced by an order of magnitude in assessing the dose against the existing onsite worker guidelines to account for this evidence of a "no-melt" scenario outcome in the frequency analysis.

Direct Dose

Consideration of "bare," unmoderated events in this analysis is not necessary, given the processes, storage conditions, and earlier hazards analyses performed for the RBOF facility. Therefore, the direct dose from the neutron and photon yield to a receptor on the traveling bridge has been calculated for minimum water and credible water levels (Ref. 78).

The basis fission yield magnitude is $5E+17$ fissions, a fission yield that is exceeded less than 5% of the time based on the distribution of relevant data drawn from accidental criticality events and experimentation (Ref. 92, 93). The prompt direct dose from this size criticality is approximately 12 rem, assuming a minimum distance between the source and receptor, and a minimum water height condition (Ref. 78). For credible water depths of about 7 feet, the direct dose is substantially less, 0.6 rem (Ref. 94).

Shine and Inhalation Doses

The assumptions and results for the inhalation and shine contributions to the dose to the facility receptor are summarized here. The total dose varies depending on the set of assumptions applied.

Instantaneous Release

The shine and inhalation doses are a function of the radioactivity released following the criticality event and the "stay time" of the facility receptor. Under worst-case conditions, the most limiting situation would be one in which noble gases and volatile fission products are released instantaneously to the basin water. Radioactive material exists prior to the event that

represents long-lived products in the RFA. The other radioactive release is from "new" fission products formed during the hypothetical event. However, the low volatility radionuclides (e.g., strontium and plutonium) would not be released to the water in the same early time frame of interest.

Some scrubbing of halogen and particulates occurs based on 7 feet of water above the criticality, assuming no steam and/or water displacement, but much of the released material is released from the basin water and is assumed mixed in the air space directly above the basin area and is thus subjecting facility workers to additional radiological components after the direct dose condition ends. With Decontamination Factors (DFs) of 100 and 1,000 for the halogens and particulates, respectively, the immersion and inhalation dose is 6.3 rem/min (Ref. 95). This radiation level would impact radiation alarms (NIMs, GA-6 monitors, and in time the stack beta/gamma monitors). Assuming no ventilation (radiation alarm/ventilation interlock), the facility receptor would have 3.9 minutes to evacuate before incurring a total dose of 25 rem and more than 15 minutes before incurring a total dose of 100 rem.

Undetected, Long Slow Release - With Ventilation

The source term for this condition represents a 60-minute fission product decay time, and the release is over the same 8-hour period as considered in the BIO (Ref. 46), based on the NRC Regulatory Guides (Ref. 96, 97). In other words, the fission products are released over 8 hours and subject the facility receptor to shine and inhalation doses after the direct (neutron and photon) dose has been incurred. It is assumed that the radiation alarms have not been triggered. Thus, the ventilation system continues to operate and the worker is not aware that the criticality event has occurred. The level of fission products is "decayed" with a 1-hour cooling time assumption, and DOE-HDBK-3010-94 (Ref. 75) release fractions are used.

The base calculation for this case is described in S-CLC-H-00492 (Ref. 98). The source term has two components for analysis purposes: a shine component made up of the krypton and xenon noble gases and an inhalation component made up of halogens, particulates, and other nonnoble radionuclide groups. The shine component is dominated by short-lived fission gases that are formed from the criticality event. The inhalation source term includes fission products from the fission event and residual radioactivity from the RFA. However, the two components will be analyzed for individual dose rates and combined for the overall dose rate.

The following assumptions apply:

- The ventilation system exhausts 16,950 cfm from the RBOF area. The effect of ventilation is assessed as a decay term using the referenced ventilation data to establish a ventilation constant of ~ 4.9/hr. Radionuclides under consideration are uniformly mixed in the available volume.
- The shine component of noble gases (krypton and xenon) is released to the building immediately after the criticality event ($t=0$). The inhalation component is released linearly over an 8-hour period, implying a constant release rate.

- Preliminary results show that the doses are singularly dominated by the ruthenium component. Review of the source documentation (Ref. 99) shows that, in the fuel type and reactor considered, this radionuclide is assumed to have appreciable release through overlying water cover. As a particulate, this assumption is considered overly conservative and not applicable to most of the fuels contained in the RBOF. Two cases will be presented. The first case is a bounding case, using the source term suggested by DOE-HDBK-3010-94 without modification. The second case will be to assume that the ruthenium is decontaminated by in-pool transport as any other particulate, justifying a 5E-04 Leak Path Factor (LPF) as applied to the other particulates. The unmodified ruthenium source term uses an ARF of 2E-03 and an LPF of 1.0, providing a 2E-03 source reduction factor. The modified source term uses the same 2E-03 ARF but applies the 5E-04 LPF, providing a 1E-06 source reduction factor.
- A DOE occupational breathing rate of 1.2 m³/hr (Ref. 100) is used.

The approximation analysis for both source term cases shows that the 25-rem and 100-rem doses are reached (conservatively) after 15 and 40 minutes, respectively, for the bounding case. The best-estimate case does not reach the 100-rem dose over the 8-hour release model. The cumulative dose after 2.5 hours is 25 rem, and the cumulative dose after 8 hours is ~80 rem. Further inspection of this analysis shows an increase to ~20 rem/hr in the first 3 minutes. This increase is the effect of the shine gases being assumed immediately released into the RBOF volume. The shine effect is rapidly diminished in the next 30 minutes as the ventilation system expels the shine component.

The bounding case, using the unmodified "Mishima" source term, indicates that the shine and inhalation doses from a process-induced criticality involving 5.0E+17 fissions would sum to 95 rem after approximately 40 minutes. Including the direct dose of ~1 rem from neutron and photon radiation from the fission event itself indicates an overall dose of ~96 rem to the worker closest to the event.

A best-estimate case, with reduced ruthenium contribution, indicates that the shine and inhalation doses of ~77 rem would be incurred after 8 hours. The overall dose then would be approximately 78 rem, when the 1-rem direct dose is included.

Undetected, Long Slow Release - Without Ventilation

The same conditions are repeated as summarized in the previous section on releases with ventilation. However, the undetected criticality is coupled with loss of ventilation. These independent events clearly comprise an overall incredible situation. It is analyzed to show the sensitivity of the radiological outcome to the facility receptor based on the ventilation system's status.

Under the same DOE-HDBK-3010-94 conditions as above, the 25-rem and 100-rem limits are reached in 1 hour and 2 hours, respectively, for the best-estimate ruthenium release assumption

(Ref. 101). The worst-case ruthenium release case for this same set of assumptions indicates the 25-rem and 100-rem limits are reached in approximately 14 minutes and 28 minutes, respectively.

Summary of Doses

The direct, inhalation, and immersion doses discussed above are tabulated in Table 3.3-18. The most appropriate combination of fission yield, source term assumptions, radiation alarm status, and building ventilation status conditions for facility worker consideration would be the credible water level direct dose of 0.6 rem, coupled with either the instantaneous release conditions (25 rem in ~4 minutes [assuming no credit split fraction]), or the long-slow release with building ventilation and without split fraction best-estimate ruthenium conditions (25 rem in approximately 2.5 hours). Other combinations are extremely low probability combinations of physics, radiation alarm status, ventilation conditions, and source term assumptions.

A second ventilation case is shown with full-facility ventilation rates and volumes included as a sensitivity to the ventilation constant. There is no appreciable difference since ventilation constants are virtually the same, or 4.9/hr in the main basin area case versus 5.2 for the full-facility purge case (Ref. 102).

Hazardous Chemical Worker Safety

Specific safety practices at the RBOF and RRF include provisions for handling hazardous chemicals. Operating procedures specify the use of special protective items when handling hazardous chemicals. These items include acid goggles, acid gloves, and an acid apron when working with nitric acid, caustics, sodium nitrite, oxalic acid crystals, and sodium dichromate crystals. These safety measures are used when disconnecting portable deionizer hoses and charging resin to the process or when removing the resin from the process. A full acid suit or plastic suit is required for line breaks that contain acid or caustic. This activity includes disconnecting acid or caustic tanks on chemical pad No. 1. Design features that contribute to worker safety include the chemical pad berms and drains that remove any spills to the waste cell. Other administrative features to ensure worker safety include the following:

- Industrial Hygiene Program
- Control of flammable antifreeze in waste tank EP38
- Removal of flammable vapors from waste tank EP38

General administrative programs and policies at the RBOF and RRF that contribute to worker safety include the following:

- Fire Control Program
- Emergency and fire response team

- Spill control team
- RC&HP response
- Onsite medical unit
- SRS safety manual and training
- Inspections and assessments
- Flushing of high- and low-activity drain lines
- Signs and postings
- SRS emergency preparedness plan
- Procedures for welding, line break, spill control/containment, and scaffolding/rigging
- RBOF/RRF general safety rules
- 100-ton crane operation safety rules
- Operation of telescoping portable platform scaffold safety rules
- Filter aid preparation area safety rules
- Building 244-H carport and cask wash pit safety rules
- Continuing occupational health training

Several administrative and operational features will be assigned TSR coverage in the form of administrative controls for adequate worker safety. These features include the following:

- Hazardous Inventory Material Control Program
- Nitric acid and Sodium Nitrite Storage Tanks
- Chemical pad No. 2 berm

The worker safety features described in this section are an integral part of the facility design and operation. These worker safety features are adequate and have also been shown to be effective through many years of safe operation at the RBOF and RRF.

Worker safety is ensured by the design features and administrative features that contribute to defense in depth. Section 3.3.2.3 discusses these RBOF/RRF features. These RBOF/RRF design features and administrative features protect the public, site worker, and the environment. Thus, each of the design features and administrative features contributing to defense in depth also contributes to worker safety. Specific programmatic safety procedures that contribute to worker safety include a fire protection program, an area emergency plan, and a radiological protection program. As discussed previously in Section 3.3.2.3, administrative procedures are in place to ensure safe operations. Chapters 6 through 12, 14, and 15 discuss in detail these programmatic safety procedures and operating procedures.

Worker safety is generally controlled through established process safety management programs as previously described. To determine the potential SSCs required for reducing the consequences of the identified accident scenarios, analysis has been performed for the primary scenarios of concern in the RBOF. From the hazardous chemical perspective, onsite consequence analysis was developed specifically for the nitric acid tank failure scenarios.

Section 3.4 discusses the use of Emergency Planning Hazards Assessment (EPHA) analysis in SARs. In summary, although not as bounding as the analysis performed for the determination of offsite consequences, the EPHA is considered acceptable for use in determining potential consequences for onsite receptors to facilitate functional classification and TSR development. This analysis (Ref. 103) confirmed that for an unmitigated nitric acid tank spill, consequences for onsite workers at 30 meters are below the ERPG-2 and -3 values. Consequently, no mitigative SSCs are required to reduce air concentrations to below the onsite EGs specified in WSRC Procedure Manual E7 (Ref. 12).

ENVIRONMENTAL PROTECTION

The primary environmental concern with respect to the RBOF and RRF is the release of certain radionuclides to the environment. The hazard evaluation documented in Section 3.3.2.3 identifies release pathways in which radionuclides could be released and adversely impact the public, site worker, and the environment. The potential for large material releases is minimized by the design and administrative features that contribute to the defense in depth of the RBOF and RRF. These design features and administrative features protect the public, site worker, and the environment and are discussed in the Defense-In-Depth section. It is appropriate to note that although there have been no specific accident EGs developed for the environment, protection of the public and site worker provides de facto protection of the environment. Thus, each of the design and administrative features that contributes to defense in depth also contributes to environmental protection.

Additional administrative and design features not included in the defense-in-depth discussion contribute to environmental protection. These administrative and design features include the following:

- Limits on releases external to the building inherent in the EGs for protection of the public (administrative control)
- Numerous legally enforceable restrictions on normal operational releases (administrative control)
- Notification and response requirements for releases of RQs in 40 CFR 302.4 (Ref. 29) (administrative control)
- Engineering good practices (administrative control)
- Location of the RBOF and RRF in an area where the topography is conducive to minimize the possibility of water intrusion to the building (design feature)

As a result of the above administrative and design features and the defense-in-depth discussion, design and operational controls are in place to prevent large releases that could cause significant environmental damage.

ACCIDENT SELECTION

The accident analysis requires a formal quantification of accidents that represent a complete set of bounding conditions. The identification of these accident scenarios results from the hazard evaluation ranking of the complete spectrum of facility accidents, as provided in the hazard evaluation tables (Tables 3.3-13 through 3.3-17). The hazard evaluation process for the RBOF and RRF identified 135 potential events involving hazards and hazardous materials present in the facility. Table 3.3-11 provides a risk rank summary of the accident scenarios identified in the hazard evaluation tables. Section 3.3.2.3 discusses these potential events, which received a risk ranking.

Five of the 136 potential events identified in the hazard evaluation tables require quantitative accident analyses. All of these events result in a risk rank of 4. These events involve hazards that could cause loss of life or life-threatening injury to the facility worker.

In addition, a representative, bounding analysis will be performed for all hazards with a risk rank of 3 or 5. A bounding analysis is to be performed for all "accident types" of concern, unless the accident type is bounded by a unique accident analysis. In addition, the risk rank 7 events are candidates for representative bounding analysis. The postulated accident scenarios identified in the hazard evaluation tables are grouped by the event categories as defined in Section 3.3.1.3. Accident types can be defined as subgroups of these event categories.

WSRC Procedure Manual 11Q (Ref. 3) has defined the following accident types:

- Earthquake
- Winds
- Lightning
- Temperature extremes
- Rain
- Surface vehicle impact
- Aircraft impact
- Hoist, crane, or rigging failure
- Structural failure
- Adjacent facility accidents
- Missiles

- Nuclear criticality
- Explosions
- Fire
- Uncontrolled chemical reactions
- Active loss of confinement (overflow)
- Passive loss of confinement (e.g., leaks, spills)

Personnel radiation exposure was identified as an additional accident type appropriate for the RBOF and RRF and was added as an accident type. Postulated accident scenarios that fall within a particular "accident type" require individual or bounding quantitative analysis in Section 3.4 if warranted by a sufficiently high risk rank. Table 3.3-10 provides a distribution of events according to the risk rank and accident type.

Events 51, 310, 314a, 316, and 318

The following risk rank 4 events were determined to require individual, quantitative accident analyses in accordance with DOE-STD-3009-94 (Ref. 2):

- Events 316 and 318: The RRF uses 50% sodium hydroxide and 50% nitric acid to regenerate anion and cation resin, respectively. According to the RBOF HAD (Ref. 13), 1,000 gallons of each chemical are stored on chemical pad No. 1. For conservatism, the assumed inventory is 110% of the actual amount (1,100 gallons). These chemicals were evaluated and found to require a hazard assessment.
- Event 314a: According to the chemical inventory and administrative controls, 900 gallons of sodium nitrite can be stored at chemical pad No. 2. This chemical was evaluated and found to require a hazard assessment (Ref. 13).
- Event 310: During resin regeneration activities at the RBOF and RRF, the possibility exists that resin could be leaked or spilled and allowed to dry out and ignite. The resulting fire could result in an airborne release that may result in a moderate consequence to the onsite worker.
- Event 51: This event postulates a violent reaction between offsite fuels containing sodium or other materials which react with the basin water if the aluminum storage container leaks, and requires additional analysis.

Events 4, 5, 5a, 6, 7, 7a, 8b, 10, 10a, 11, 11a, 12, 13, 14, 22, 23, 25, 59, 59a, 334, 400a, and 402

The following risk rank 3, 5, and 7 events require a bounding analysis for each accident type to the extent they are not bounded by the unique, quantitative accident analysis. Industrial type hazards that impact the facility worker are not included; the Worker Safety section addresses these events. Events in which an NPH directly causes injury to the facility worker are excluded

in accordance with Reference 55. These 22 events involve facility worker radiation exposure that could result in a low consequence due to direct exposure or contamination:

- Event 4 is caused by a breach of a shipping cask discovered during a radiation survey or receipt inspection upon the arrival of the cask in the RBOF carport.
- Events 5 and 5a result from an airborne release when an offsite cask or SRS cask is dropped.
- Event 6 results in an airborne release through the building exhaust system during cask venting.
- Event 7 results in an airborne release due to fuel breach during handling. Event 7a involves target fuel assemblies that could breach during handling or movement in the RBOF basins.
- Event 8b involves exposed fuel during an inadvertent basin draindown.
- Events 10 and 10a involve an airborne release occurring as the result of inadvertent cutting of fuel during cropping or inadvertent cutting of a target core during cropping in the disassembly basin, respectively.
- Events 11 and 11a involve an airborne release occurring when fuel or target assemblies breach while in the storage basins.
- Events 12 and 13 result in an airborne release during cask cooling through the building exhaust system and between the cask and desuperheater, respectively.
- Events 14 and 22 involve the exposed movement of fuel (i.e., either removing fuel from the cask or raising the fuel above the basin water level). In these cases, response to area radiation alarms would limit the radiation dose to the facility worker.
- Event 23 postulates that a damaged cask received by rail or truck could result in the radioactive contamination of workers.
- Event 25 involves receipt of a fuel shipment with an erroneous shipping manifest, resulting in unspecified radiation doses to the workers.
- Events 59 and 59a involve worker contamination or radiation exposure during maintenance activities.
- Event 334 postulates a breach of deionizers or associated piping, hoses, or valves that could cause radiation exposure to workers during resin transfer operations.
- Event 400a postulates that a loss of containment occurs due to energetic reaction caused by mixing undiluted nitric acid and cation resin during resin regeneration, resulting in an injury to the facility worker.
- Event 402 postulates worker radiation exposure due to a loss of containment of the liquid, low-level waste trailer drums.

These events only affect facility workers and are primarily addressed in Section 3.3.2.3.

Events 34, 35, 37b, 38, 39, 40, 40a, 41, 42a, 43, 44, 63, 63a, and 64

Because of the large amounts of fissile material present in the form of irradiated fuel assemblies and the presence of water for moderation, criticality is possible in several areas of the RBOF basins. The postulated criticality can occur through process upset conditions (i.e., internal event initiators) or through NPH-induced conditions. The latter class of events includes seismic events, tornadoes, and high winds. The risk rank 3, 5, and 7 events are as follows:

- Event 34 postulates a criticality occurring as a result of a floating fuel container. This event could be caused by a storage violation, an equipment malfunction, or a defective container.
- Event 35 postulates a criticality occurring as a result of a fuel storage infraction. This event could be caused by a procedure violation or worker error.
- Event 37b postulates a criticality occurring as a result of a deformation of the fuel storage rack. The fuel storage rack deformation in this scenario is assumed to be caused by either a worker error causing the hoist to snag the rack, or dropped fuel.
- Event 38 postulates a criticality occurring as a result of a critical configuration in a fuel bundle. This event could be caused by worker error in assembling a fuel bundle.
- Events 39 and 44 postulate a criticality caused by fissile debris on the basin floor as a result of fuel breach or damage.
- Event 40 postulates a criticality occurring when fuel spills from the basket, and Event 41 postulates a criticality occurring when more than one basket is out of the cask at the same time.
- Event 40a postulates a criticality occurring when the contents of a shipping cask are spilled into the basin or the cask lid is dropped back into the cask.
- Event 42a postulates a criticality occurring due to draining the 70-ton cask.
- Event 43 results when the water level in the basin leaks into the subsoil as a result of an earthquake, causing decay heat to deform the fuel and resulting in criticality.
- Events 63, 63a, and 64 are NPH events that cause damage to the fuel stored in the basin, resulting in a criticality.

There are two major consequence concerns from nuclear criticality events: direct dose from prompt gamma and neutron radiation and dose from plume immersion (cloudshine and inhalation pathways). The total dose is the sum of the direct dose and the plume immersion dose evaluated at each receptor distance. The consequences of a criticality event could result in a radiological impact to the facility worker and to the onsite worker. The MOI may not be impacted by these events. Therefore, criticality events require additional analyses that are bounding for these event types.

Events 314b, 315, 317, 328c, 345c, and 345d

These risk rank 3, 5 and 7 events are all concerned with hazardous chemical exposures to the facility worker, resulting in a serious injury.

- Event 314b involves a release of less than 167 gallons of 30% sodium nitrite on chemical pad No. 2.
- Event 315 involves a release of less than 72 gallons of 50% sodium hydroxide that occurs on chemical pad No. 1.
- Event 317 involves a release of less than 71 gallons of 50% nitric acid that could occur on chemical pad No. 1.
- Event 328c involves a release of less than 30,750 gallons of ethylene glycol.
- Events 345c and 345d postulate a ground vehicle crashing into chemical pad Nos. 1 and 2, respectively, that could result in a breach of containment of chemicals stored there.

Therefore, hazardous chemical spill events require additional analyses that are bounding for these event types.

Events 347, and 348

These risk rank 3, 5 or 7 events, 347 and 348 are NPH events that cause damage to the chemical pads or hose racks resulting in a release of hazardous chemicals and radionuclide contaminated water.

Event 3

Event 3 postulates an explosion and/or fire with low consequences. Event 3 postulates a minor explosion that occurs during cask venting when a mixture of hydrogen, oxygen, and air in the cask explodes, resulting in a low consequence.

3.4 ACCIDENT ANALYSIS

The accidents analyzed in Section 3.4 are derived from the hazards analysis (Section 3.3). Potential accident events identified in the hazards analysis were binned into risk categories based on likelihood of occurrence and consequence. Each risk bin is assigned a risk rank in accordance with DOE-STD-3009-94 (Ref. 2). The standard states that a unique accident analysis should be performed for all hazards that are a major concern in the hazards analysis (risk rank 1, 2, or 4), and a representative bounding analysis should be performed for all hazards that are a marginal concern in the hazards analysis (risk rank 3 or 5). Risk rank 7 events are only considered as candidates for the representative bounding analysis. A bounding analysis is performed for all accident types of concern, unless the accident is bounded by a unique accident analysis. Accidents analyzed in Section 3.4.2 include the following, as defined by DOE-STD-3009-94:

Unique Accidents

- Chemical spills
 - Loss of 50% sodium hydroxide containment (Event 316)
 - Loss of 50% nitric acid containment (Event 318)
 - Loss of 30% sodium nitrite containment (Event 314a)
- Portable deionizer resin fire (Event 310)
- Uncontrolled chemical reaction
 - Violent reaction between the sodium fuel and water (Event 51)

Representative Bounding Accidents (note that in some cases, the accident type is bounded by the events analyzed above as unique)

- **Earthquake** - Event 63a (deformation of the fuel storage rack with basin draindown) and Event 43 (low water level in the basin) were both chosen for analysis due to the potential differences in risk. Each event propagates into a criticality accident.
- **High Winds/Tornado** - Events 63, 63a and 64 (deformation of the fuel storage rack) resulting in a criticality.
- **Severe Weather** - Event 348 (damage to the chemical pad and hose racks due to temperature extremes, snow, hail, or ice) results in a containment breach.
- **Surface Vehicle Impact** - Events 345c and 345d are bounded by the three unique chemical accidents: Events 314a, 316, and 318.
- **Aircraft Impact** - This accident will be shown to be incredible, and therefore no accident scenario needs to be developed.
- **Structural Failure** - Events 63, 63a and 64 (deformation of the fuel storage rack) due to earthquake, hurricane (straight winds), or tornado resulting in a criticality.

- **Missile Impact** - This accident is bounded by the three unique chemical accidents: Events 314a, 316, and 318.
- **Criticality** - Process-related criticality is hypothesized for several scenarios: Events 34, 35, 37b, 38, 39, 40, 40a, 41, 42a, and 44. The most likely and energetic is discussed here.
- **Explosion** - Event 400a (addition of undiluted nitric acid to cation resin during resin regeneration) causes an explosive reaction.
- **Fire** - Event 310 (a resin fire) results in a radioactive airborne release at ground level.
- **Uncontrolled Chemical Reaction** - This accident is bounded by unique Event 51.
- **Active Loss of Confinement** - This accident is bounded by the three unique chemical accidents: Events 314a, 316, and 318.
- **Passive Loss of Confinement** - This accident is bounded by the three unique chemical accidents: Events 314a, 316, and 318.

Of the risk rank 7 events listed in the accident selection of Section 3.3, Event 400a was considered to be the representative bounding analysis due to the Material At Risk (MAR).

In accordance with DOE Order 5480.23 (Ref. 1), the accidents or events that are quantitatively assessed are grouped into three general types for purposes of discussing them in the SAR: operational (or process-related) events, NPH events, and external events.

The graded approach has been employed in this section by (1) "fine-tuning" accident analyses only to the point where the assessment confirms that the calculated consequences of the event are within EGs, which minimizes the effort spent on any particular analysis, and (2) performing bounding consequence assessments (i.e., using bounding inventories for the subfacilities and containers involved in the postulated accidents), which minimizes the total number of analyses that are performed.

The description for each quantitative accident scenario relates the event to one or more of the specific Design/Evaluation Basis Accident Types that have been defined for Nuclear Facility Unreviewed Safety Question Evaluations (USQEs) in the evaluation procedure (Ref. 3), and Table 3.4-12 of WSRC-TR-94-0466 (Ref. 56). WSRC-TR-94-0466 is a WSRC document providing additional guidance (i.e., in addition to DOE-STD-3009-94 [Ref. 2]) on preparation of SARs defined by DOE Order 5480.23 (Ref. 1).

3.4.1 Methodology

This section describes the methodology used for the accident analysis of the RBOF.

3.4.1.1 Scenario and Source Term Development

The basic methodology used for determining the bounding amount of hazardous material released from a facility to the atmosphere is based on techniques described in DOE-HDBK-3010-94 (Ref. 75), which summarizes the experimental data related to airborne radionuclide releases. The quantity of released, airborne, respirable material is referred to as the Source Term (ST), which is defined by one of the following equations:

$$ST = MAR * DR * ARF * LPF * RF, \text{ or} \quad (\text{Eq. 3.4-1})$$

$$ST = MAR * DR * (ARR * t) * LPF * RF \quad (\text{Eq. 3.4-2})$$

where:

MAR = Material At Risk - The amount of radioactive or hazardous material present in a discrete physical location that is separated from other radionuclide or hazardous material-bearing areas by adequate distance or physical barriers so that the areas do not interact during postulated accident conditions.

The units of MAR are discussed later in this section. The next six factors are ratios or fractions; they have no units (except for ARR and t).

DR = Damage Ratio - The fraction of MAR impacted by the accident-generated conditions.

ARF = Airborne Release Fraction - The coefficient used to estimate the amount of a radioactive material that can be suspended in air and available for transport under a specific set of accident conditions.

ARR = Airborne Release Rate - The coefficient used to estimate the amount of a radioactive or hazardous material that can be suspended in air (per hour) by continuously acting mechanisms such as aerodynamic entrainment/resuspension.

LPF = Leak Path Factor - The fraction of airborne materials transported from containment or confinement by the existing flow via the pathway configuration under evaluation.

RF = Respirable Fraction - The fraction of airborne particles that can be transported through air and inhaled into the human respiratory system. These particles are commonly assumed to be 10 μm aerodynamic equivalent diameter and less. (Aerodynamic equivalent diameter is the diameter of a sphere of density 1 g/cm^3 that exhibits the same terminal velocity as the particle in question.)

t = Release duration time (hours)

The selection of one of the two equations given above for the source term analysis of a specific event is based on the type of event and the nature of the release data available to characterize it. Events where the release occurs suddenly generally employ Equation 3.4-1. Events where the release occurs over an extended time, such as an evaporative release from an open pool of volatile liquid resulting from a spill, generally employ Equation 3.4-2.

The most important factors in the above equations are the DR and the combination of $ARF \times RF$ (or $ARR \times t \times RF$). The ARF, ARR, and RF are generally obtained from DOE-HDBK-3010-94 (Ref. 75).

The LPF is set to 1.0 for all RBOF events with the exception of the criticality events. The LPF for these events account for the source term reduction due to the basin water.

It should be noted that not all of the factors mentioned above appear in the source term equation for each event. When the factors are not listed, a value of 1.0 is assumed.

Following the discussion dealing with various factors comprising the basic method, several sections cover other aspects of the source term analysis and related topics. These include a discussion of the treatment of releases from Hazard Category 2 subfacilities and a discussion of how the source terms associated with identification of candidate SSCs for functional classification are developed.

MATERIAL AT RISK (RADIOACTIVE AND HAZARDOUS CHEMICAL INVENTORIES)

The inventories used in the source term analyses are listed in corresponding accident analysis sections. Additional information about these inventories is contained or identified in WSRC-TR-94-054 (Ref. 13 and 59), which is a compilation of RBOF inventories approved by the facility engineering organization for use in safety analyses. These inventories have been generated for use in safety assessments and are generally conservative with respect to (i.e., larger than) the actual (nominal) values for facility inventories. In particular, some administrative controls or operational restrictions have not been credited in determining the MARs used in the accident analyses.

AIRBORNE RELEASE FRACTIONS AND RESPIRABLE FRACTIONS FOR RADIONUCLIDE RELEASES

The ARFs of DOE-HDBK-3010-94 (Ref. 75) are used where applicable. Section 3.4.2 discusses the specific ARF and RF values employed for various types of events (i.e., postulated accidents) and lists the specific sections within DOE-HDBK-3010-94 describing the basis for these ARF and RF values.

AIRBORNE RELEASE FRACTIONS FOR CHEMICAL RELEASES

Chemical release source term analyses are performed using methods that are similar to those employed for radioactive nuclides. The key difference is that the respirable release fraction (the product ARF x RF) employed for chemical releases is related to the volatility of the chemical.

For volatile and semi-volatile compounds (which includes most liquid hazardous chemicals), the analysis uses a 1-hour release fraction obtained from a computational method based on the analogy between mass transport by convection and heat transport by convection. This method (sometimes referred to at SRS as the Eckert-Drake model), which is described below, is adapted from NUREG-0570 (Ref. 105) and attributed to Reference 106.

The method assumes mass transfer by evaporation under isothermal conditions, with both the liquid and the surrounding air at 35°C. The liquid is generally assumed to have formed a 1-cm-deep circular pool (unless a structure, such as a sump or berm, is credited to limit its spreading), and the evaporation is driven by a 4.5-m/sec air flow (outdoors) over the pool.

The liquid volume is calculated from the mass of the chemical inventory at risk and the density of the spilled liquid. The pool area is calculated by dividing the liquid volume by the (assumed) 1-cm pool depth, and the diameter is calculated from the calculated area for the (assumed) circular pool.

The mass transfer rate is determined by the following equation:

$$\dot{m} = hA \frac{M\Delta p}{RT} \quad (\text{Eq. 3.4-3})$$

where:

A	=	pool surface area
M	=	molecular weight of the volatile compound
R	=	universal gas constant
T	=	air temperature
Δp	=	difference between liquid-phase vapor pressure and gas-phase partial pressure
h	=	mass transfer coefficient

The mass transfer coefficient (h) is determined from the following equations:

(laminar)

$$h = 0.664 \frac{D}{d} \text{Re}^{\frac{1}{2}} \text{Sc}^{\frac{1}{3}} \quad (\text{Eq. 3.4-4})$$

(mixed)

$$h = \frac{D}{d} (0.037 \text{Re}^{0.8} - 871) \text{Sc}^{\frac{1}{3}} \quad (\text{Eq. 3.4-5})$$

(turbulent)

$$h = \frac{D}{d} (0.037 \text{Re}^{0.8}) \text{Sc}^{\frac{1}{3}} \quad (\text{Eq. 3.4-6})$$

where:

- d = pool diameter
- D = diffusion coefficient of the volatile compound in air
- Re = Reynolds number
- Sc = Schmidt number

The Reynolds number and Schmidt number are defined as follows:

$$\text{Re} = \frac{du\rho_a}{\mu_a} \quad (\text{Eq. 3.4-7})$$

$$\text{Sc} = \frac{\mu_a}{D\rho_a} \quad (\text{Eq. 3.4-8})$$

where:

- u = wind speed
- ρ_a = air density
- μ_a = air viscosity

The transitional Reynolds number between laminar and mixed flow is 5E+05.

The release fraction is determined by dividing the mass released over 1 hour by the total mass inventory available for release.

Accident scenarios involving chemical dispersion by more than just evaporation (e.g., splashing) are discussed in the applicable accident analysis portion of Section 3.4.

CREDITING PROCESS EQUIPMENT

This section describes preventive and mitigative features for the postulated accident scenarios. For each scenario, several preventive and mitigative features are listed and described briefly. If the feature was credited for the accident scenario, it is annotated. Otherwise, it is listed as a defense-in-depth feature.

ASSUMPTIONS

This section describes assumptions dealing with physical phenomena and facility operations characteristics. These assumptions are exclusive of, and in addition to, the operations characteristics described in the preceding section.

3.4.1.2 Frequency Assessment

The frequency at which process accidents, process upsets, or system failures occur must be quantified to facilitate risk evaluation and to aid in the selection of TSRs and safety-class/safety-significant SSCs. General frequency methodology is described below, followed by a discussion of the specific methodologies used to establish the frequency for each analyzed accident. Frequency estimates resulting from these methods are generally probabilistic and should not be treated as deterministic values or used as such in a deterministic calculation.

GENERAL FREQUENCY METHODOLOGY

Estimation of High Frequency Events

When historical data on events are available, it is generally preferable to use those data to estimate the frequency of event occurrence, rather than more complex inductive methods required when applicable data are not available. For those process-related events that occur frequently, an abundance of data usually exists on both component and system failures. Site-specific data for SRS can be obtained from the various SRS data banks.

If data are not available on the specific system of interest, they may be obtained from safety analyses for similar facilities or from generic sources. If differences are known to exist between the two systems, adjustment of the estimate may be advisable to ensure an appropriate level of conservatism. Generic sources may vary greatly in quality and quantity of information provided. Thus, data should normally be drawn from well-established, defensible databases. At SRS, the

two most common sources of generic data are WSRC-TR-93-581 (Ref. 60) for human error data and NUREG/CR-4639 (Ref. 107) for equipment failure.

Estimation of Low Frequency Events

Occurrence frequency estimation for events that occur very infrequently tends to be more complex due to the small amount of data and corresponding need for failure model development. Two popular methods of developing frequencies or probabilities of failure for this category are fault tree analysis and event tree analysis. Of more limited applicability is the method based on Rare Event Theory.

FAULT TREE ANALYSIS

Qualitative determination of the detailed combinations of events that can lead to an accident and subsequent quantification of the associated frequency of occurrence can be effectively accomplished using fault tree analysis. Fault tree analysis is an analytical technique by which an undesired state of the system is specified, and the system is then analyzed to find all possible ways in which the undesired state can occur. A fault tree is a graphic model of the logic representing various parallel combinations of faults that result in the occurrence of the predefined event. The faults can be events associated with component hardware failures, human errors, or any other pertinent events that can lead to the undesired event.

Model Development

A detailed discussion of the development of fault tree models is presented in NUREG-0492 (Ref. 108).

Data Development

Quantification of fault tree models is dependent on the application of credible data. A number of suitable data sources are readily available. Developed primarily for power reactors, NUREG/CR-4639 (Ref. 107) provides a Nuclear Computerized Library for Assessing Reactor Reliability, much of which can be applied to nonreactor facilities. WSRC-TR-93-262 (Ref. 120) is a compendium of both SRS failure data and failure rate estimates for more than 500 component / failure mode combinations associated with water, chemical, gas, HVAC, electrical, instrumentation, and control systems. IEEE-500 (Ref. 109) provides similar data for electrical and electronic components. WIN-330 (Ref. 110) contains failure rate data on components for the Idaho facility, much of which is of general utility.

Human errors can act as accident initiators and propagators. Human errors frequently are more important in the accident analysis than equipment failures. Generic data on human error probabilities are provided in WSRC-TR-93-581 (Ref. 60).

Model Evaluation

The fault trees are generally created and evaluated using the Computer-Aided Fault Tree Analysis (CAFTA) computer program (Ref. 111). CAFTA is a computer software tool used to create, modify, and evaluate fault tree models. It includes a fault tree editor, a basic event database editor, formatting routines, fault tree analysis routines, and a cutset editor. CAFTA allows the input of failure rate data, mission times, and error factors for uncertainty calculations. It also allows the user to input the appropriate relationships between system components to evaluate a system failure frequency or probability of failure.

The principal CAFTA interface is the fault tree editor, which is used to create and modify fault trees. Syntax and logic verification is accomplished interactively as operations on the models are being performed. The fault tree editor also links basic event reliability data for multiple trees into a single database file. This basic event database includes the basic event name, the trees in which the event is used, information on methods for calculating the failure probability, the data source, and the mission/repair time.

Basic event inputs to the fault tree models are classified as one of three types: initiators, enablers, or initiator-enablers. An initiator failure may directly result in a facility accident if a redundant or supporting system is off line at the same time. The off-line system is said to have failed as an enabler because its unavailability allows the accident to occur if an initiator has failed. There are many situations in which a single component can serve either as an initiator or as an enabler. To ensure the proper expression of both initiator-enabler modes in the fault tree solution, the IBOTH code is used in the system analysis quantification (Ref. 112).

A final consideration involved in the development of the fault tree is the treatment of common-cause failure. Common-cause event modeling is performed to relate groups of two or more components associated by design, maintenance, system, or function. Group failure rates are obtained using the Beta method, in which multiplicative factors are applied to individual component failure rates to determine group failure rates (Ref. 113, 114). A beta value of 0.1 is used for groups of two similar components, and a beta value of 0.05 is used for groups of three or more components.

Following fault tree model development and basic event data input, the CAFTA fault tree quantification processor, CQUANT, is employed to obtain algebraic expressions of basic event failures that reflect the combinatorial logic of the model. The combinations of component failures that are numerically added in this top event expression, referred to as fault tree cutsets, represent the groups of failures that can result in top event occurrence. Numeric evaluation of the fault tree cutsets results in the estimation of accident initiation and support system failure frequencies.

EVENT TREE ANALYSIS

The event tree is an analytical tool that allows the analyst to organize and characterize potential accident sequences and provide a graphical logic model for quantifying outcomes of initiating events. The benefit of developing an event tree logic model is that it can provide a concise overview of all postulated accident scenarios considered in the analysis. Using it as a model provides the analyst the means to describe in a comprehensive manner important aspects of the accident sequences.

Consequently, the primary use of the event tree model is to aid the analyst in presenting the spectrum of significant accident sequences, including describing their effects on the site relative to the various initiating events postulated in the analysis.

Model Development

Based upon facility design information, potential accident-initiating events are identified. Event trees are developed to depict the ability of the facility to mitigate or prevent the potential effects of these events. This comprehensive approach provides reasonable assurance that the significant causal factors affecting the safe operation of a facility have been accommodated in the analysis and that the results reflect a realistic bounding of potential accidents.

Once the initiating events have been described, functional event trees are developed. These event trees describe the functional relationships among safety- and nonsafety-related systems, operator actions, and facility behavior and place them into a binary event tree logic structure. When these functional relationships have been established, the events are expanded to depict the response of these various constituents that may mitigate the effects of a particular accident initiator. These detailed event trees define specific system and operator response requirements, which are translated into top events. Event trees are developed to determine the probability of success or failure in achieving these requirements for mitigating the scenario.

Data Development

Event trees are used in the analysis of RBOF operations to determine the frequency of an NPH-induced criticality. Mean AEPs (Ref. 115, 116) are combined with critical component and structural failure probabilities derived from RBOF capacity and uncertainty data (Ref. 117) and with failure probability estimates derived from knowledge-based judgment (Ref. 91).

Model Evaluation

Simple models can be evaluated using ETA-II or a similar program (Ref. 118). These programs determine the frequency for each sequence by obtaining the product of the probability of occurrence of each branch involved in the sequence.

RARE EVENT THEORY

For events that have not occurred in a facility and do not lend themselves to fault/event tree analysis, a method based on Rare Event Theory (as described in Section 3.3.1.3) may be used. This method is useful only if a very conservative estimate of frequency is appropriate, because this method usually overstates the frequency of an event to provide a high-confidence estimate.

It should be noted that the value of t should be very large for practical application of this method. Otherwise, the upper limit value may be overly conservative (i.e., too large to be useful) (Ref. 119).

SPECIFIC FREQUENCY METHODOLOGY

This section briefly describes the specific methodology used in estimating the frequency of each accident analyzed in the accident analysis.

Chemical Spills

A fault tree analysis is performed to quantify the frequency of chemical spills due to initiating events other than natural phenomena or external events. The frequencies of chemical spill due to natural phenomena and external events are addressed in separate sections. The logic of the fault tree accounts for all significant contributors, and the basic events in the fault tree are assigned frequency or basic event probabilities based primarily on WSRC-TR-93-581 and WSRC-TR-93-262 (Ref. 60, 120).

Surface Vehicle Impact

The frequency of surface vehicle impact is based on data from a previous safety analysis and the frequency of vehicle traffic in the vicinity of the RBOF. The data obtained from the previous safety analysis (Ref. 121) are estimations of vehicle accident frequencies per mile traveled, which are combined with estimated traffic rates in the RBOF vicinity.

Process Error-Induced Criticality

A fault tree analysis is performed to quantify the frequency of criticality at the RBOF due to initiating events other than natural phenomena. The logic of the fault tree accounts for the number of fuel handling operations performed per year at the RBOF, coupled with errors that could potentially bring a critical mass of fuel together into a geometry favorable for criticality. Conditional probabilities are added to the tree to account for whether or not the scenario would actually result in a criticality. The methodology used to determine whether or not scenarios would result in a criticality is discussed in Section 3.4.1.4. If the scenario was deemed to be capable of resulting in a criticality, a conditional probability of 1.0 is assigned, whereas if a

scenario is deemed to be incapable of resulting in a criticality, a conditional probability of $1\text{E-}06$ is assigned. Numerical values assigned to other basic events in the fault tree are based primarily on WSRC-TR-93-581 and WSRC-TR-93-262 (Ref. 60, 120).

Severe Weather-Induced Chemical Spill

The frequency of a severe weather-induced chemical spill is determined based on a simple fault tree (although not developed in the form of a fault tree), where the initiator (severe weather) is based on SRS climatological data, and enabling events are based on conservative engineering judgment. The wind-induced rain shelter failure frequency is quantified by integrating (via spreadsheet) the potential failures that result in rain shelter failure over a spectrum of possible initiating events up to evaluation basis levels. The initiating event frequencies are obtained from hazard curves for SRS, and potential failures that lead to a rain shelter failure are based on fragility estimates for components at risk.

Severe Weather-Induced Criticality

The frequency of a severe weather-induced criticality is quantified by means of event trees. The event tree sequence frequencies are quantified by integrating (via spreadsheet) the potential failures that result in criticality over a spectrum of possible initiating events up to evaluation basis levels. The initiating event frequencies are obtained from hazard curves for SRS, and potential failures that lead to a criticality and fuel breach are based on fragility estimates for components at risk and expert opinion.

Earthquake

The frequency of earthquakes is based on hazard curves for SRS (Ref. 116). The frequency of an earthquake-induced criticality is quantified by means of an event tree. The event tree sequence frequencies are quantified by integrating (via spreadsheet) the potential failures that result in criticality over a spectrum of possible earthquakes up to evaluation basis levels. The initiating event frequencies are obtained from hazard curves for SRS, and potential failures that lead to a criticality are based on fragility estimates for components at risk and expert opinion. This same approach is used to estimate the frequency of earthquake-induced rain shelter failure.

Fire

A fault tree analysis is performed to quantify the frequency of a portable deionizer resin fire. The logic of the fault tree accounts for all significant contributors, and the basic events in the fault tree are assigned frequency or basic event probabilities based primarily on NUREG/CR-4639 and NUREG-0492 (Ref. 107, 108).

Explosion

A fault tree analysis is performed to quantify the frequency of a resin explosion event. The logic of the fault tree accounts for all significant contributors, and the basic events in the fault tree are assigned frequency or basic event probabilities based primarily on NUREG/CR-4639 and NUREG-0492 (Ref. 107, 108).

Uncontrolled Chemical Reaction

A fault tree analysis is performed to quantify the frequency of an uncontrolled chemical reaction. The logic of the fault tree accounts for all significant contributors, and the basic events in the fault tree are assigned frequency or basic event probabilities based primarily on NUREG/CR-4639 and NUREG-0492 (Ref. 107, 108).

Aircraft Impact

The frequency of aircraft impact is based on data from a previous safety analysis (Ref. 122) and the "target area" of the RBOF facility. The data obtained from the previous safety analysis are estimations of aircraft accident frequencies per unit area.

3.4.1.3 Consequence Analysis

RADIOLOGICAL CONSEQUENCE ANALYSIS METHODOLOGY

Radiological consequences are calculated from postulated process-induced and NPH-induced events. Consequences are estimated for offsite and onsite receptors (Ref. 104). The offsite receptor is the MOI. The MOI is assumed to be at the site boundary and is evaluated at 99.5% worst-sector meteorology. The OEP is evaluated at 100 meters and 640 meters using median (50%) meteorology. The majority of dose calculations in the RBOF analysis to onsite and offsite receptors were calculated by the AXAIR89Q computer code (Ref. 123, 124). AXAIR89Q was retired in mid-1996. Subsequent dose calculations for the RBOF SAR analysis have been performed with the successor code, AXAIRQ (Ref. 125, 126).

AXAIR89Q

AXAIR89Q, Version 1.2, is one of the dose assessment codes used at SRS to predict downwind doses following a hypothetical atmospheric release of relatively short duration. The AXAIR89Q computer code calculates the effective dose equivalent and organ doses resulting from a hypothetical radioactive release to the atmosphere. The dispersion and dosimetry models employed in the code are those as described in Regulatory Guides 1.145 and 1.109, respectively (Ref. 127, 128). AXAIR89Q calculates doses for both vent and stack releases. Fifty-year dose

commitments to individuals and populations are determined as a result of inhalation and direct exposure ("cloudshine").

The successor to AXAIR89Q at SRS is AXAIRQ. AXAIRQ is a new, improved version of AXAIR89Q that, while still based on the guidance in Regulatory Guide 1.145, offers several substantial new features, including the following:

- Diffusion coefficients more appropriate for SRS
- Deposition/depletion option
- User-input mixing height
- 95% doses for user-selected distances

However, to be consistent with the earlier dose calculations performed with AXAIR89Q, AXAIRQ was run in a conservative, bounding mode that emulated the earlier calculations.

The AXAIR89Q code estimates ground level radionuclide concentration at given distances from a release point by using a Gaussian plume model. As the plume moves downwind, the radioactive pollutants are assumed to be normally distributed around the central axis of the plume. The diffusion coefficients representing the dispersion of the plume are those calculated by Pasquill and Gifford (Ref. 129). Plume rise from momentum and sensible energy effects is not credited in AXAIR89Q nor AXAIRQ.

The 5-year SRS database (1987-1991) is used in AXAIR89Q for statistical sampling of site meteorology. This database is quality-assured with SRS methodology to most accurately portray meteorological conditions existing at the time of the postulated release (Ref. 130). The 1987-1991 snapshot of site conditions is the most recent 5-year update available for safety analysis and has been shown to vary under some conditions with earlier (1982-1986) data (Ref. 131). The data are obtained by taking hourly averages of measurements taken at 1.5-second intervals. The data files are tower-specific with locations in A, C, D, F, H, K, and P Areas. Within these files are joint frequency distributions and reciprocal average wind velocities that are categorized by wind direction, speed, and stability.

General input parameters for the RBOF consequence analysis effort include the following:

- The H-Area meteorological tower is the sampling location.
- Releases are specified as groundlevel vent releases.
- A 2-hour release is specified for all releases (this is the AXAIR89Q-specific minimum).
- All 16 compass sectors are analyzed.
- Dose factors for converting activity to dose use the methodology of the International Commission on Radiological Protection (Ref. 132 through 135).

- Gamma-shine dose estimates are based on the upper-bound approximation of the nonuniform plume model.

Specific input parameters for each accident can be extracted from the individual Engineering Calculations that support the consequence analysis of the accident.

Radiological Consequences Comparison to Guidelines

The risks of operating the RBOF are calculated by multiplying the consequences of an accident by the frequency of the accident. The method employed to quantify this relationship is to graphically display the consequence versus frequency as a point on a "Farmer Plot," which then becomes a risk assessment point. The risks of operating the RBOF are assessed by comparing the risks of the postulated accidents against established EGs (Ref. 3). The EGs for the MOI are as follows:

- 0.5-rem limit for accidents with estimated frequencies $\leq 1\text{E-}01$ events/yr but $> 1\text{E-}02$ events/yr
- 5-rem limit for accidents with estimated frequencies $\leq 1\text{E-}02$ events/yr but $> 1\text{E-}04$ events/yr
- 25-rem limit for accidents with estimated frequencies $\leq 1\text{E-}04$ events/yr but $> 1\text{E-}06$ events/yr

Note: The 25-rem limit is based on information provided in 10 CFR 100 and DOE Order 6430.1A (Ref. 31, 136); the 5-rem limit is based on information provided in 10 CFR 72 (Ref. 31); and the 0.5-rem limit is based on information provided in DOE Order 5400.5 (Ref. 6).

Onsite receptors are evaluated against 25 rem for events with frequencies $> 1\text{E-}04/\text{yr}$ but $\leq 1\text{E-}01/\text{yr}$ and 100 rem for events with frequencies $> 1\text{E-}06/\text{yr}$ but $\leq 1\text{E-}04/\text{yr}$.

CHEMICAL CONSEQUENCE ANALYSIS METHODOLOGY

Consequences resulting from the release of chemicals from the RBOF under postulated accident conditions are limited to the airborne pathway because, as an accident scenario, this typically represents the most immediately significant route of public exposure. Air dispersion modeling uses the Areal Locations Of Hazardous Atmospheres (ALOHA) computer code and a Gaussian plume model, which is calculated by hand (Ref. 137). The heavy gas dispersion calculations in ALOHA are those used in the DEGADIS model (Ref. 138). The DEGADIS model was selected because of its general acceptance and the extensive testing that was carried out by the authors.

The hand calculation method uses the same Gaussian model and dispersion coefficients as ALOHA uses for neutrally buoyant plumes. Since ALOHA does not permit a release to last for more than 1 hour, a hand calculation facilitates the use of the appropriate plume transport times.

Under low wind speed conditions, the plume will not reach the site boundary within the allotted time. Therefore, the hand calculation is required.

The established practice for evaluating chemical concentration applies Stability Class F and a wind speed of 1.5 m/sec (Ref. 139). It is noted that within the EGs there is no discussion of an onsite receptor, so these meteorological conditions are incorporated for consideration of the offsite evaluation location only. The onsite consequences use best-estimate meteorology (e.g., 50% meteorology) in accordance with ESH-OOC-940012 (Ref. 104). This meteorology is assumed to be Stability Class D and 4.5-m/sec wind speed.

For site boundary distances from the RBOF, it is conservative to model the plume as Gaussian. Initially, a heavy gas spreads more rapidly in the cross-wind direction due to gravity. This phenomenon allows enhanced mixing as the plume transitions to neutrally buoyant behavior due to the diminishing thickness of the plume. Therefore, for a heavy gas release, use of the Gaussian plume model will overestimate the concentration at the site boundary. The Gaussian plume model is therefore conservatively used for heavy gas dispersion to the site boundary.

The salt plumes will be passively transported by the wind as particulates. Therefore, the Gaussian plume model is used for the sodium hydroxide and the sodium nitrite releases.

Use of Emergency Planning Hazards Assessment Analysis in Safety Analysis Reports

EPHAs are performed for all facilities on site that contain hazardous material (Ref. 140). The EPHA uses a deterministic barrier/failure approach for much of its accident analysis. As such, EPHA analysis would not, in all cases, be considered a "bounding" approach for accident analysis. The EPHA routinely analyzes receptor locations for the facility worker (30 meters), individual at the OEP (100 meters), and the general public (maximally exposed individual at the closest site boundary). Even though the EPHA analysis may not meet the criterion of "bounding," the analysis in an EPHA is considered very conservative.

The major concern when performing an accident analysis for a SAR is to ensure that the offsite dose is below the applicable EG. Once this criterion is met, further analysis is normally not performed to reduce the consequence because this would entail future cost. However, the use of the same bounding analysis for onsite dose calculations could burden a facility with unnecessary operational requirements. The EPHA, as a conservative analysis, may be more appropriate for use in determining potential consequences for onsite receptors to facilitate functional classification and TSR development. This is normally the case when analyzing chemical spills. Because only chemical analysis from the EPHA is used in this SAR, EPHA radiological analysis is not discussed.

Material at Risk

The MAR for the worst-case chemical spill is the maximum volume of the tank itself. The maximum volume is not always equal to the physical volume of the tank if a passive overflow is installed to limit the fill level.

Source Term Determination and Consequence Analysis

Source terms for chemicals that are in their pure state (e.g., not solutions) are calculated using the ALOHA code. The typical assumptions used are as follows (Ref. 137):

- Instantaneous spill to 1 cm depth
- 95% adverse SRS historical wind speed of 1.7 m-sec^{-1}
- 95% adverse SRS historical temperature of 29°C

The physical data for the chemical of concern (e.g., boiling point, molecular weight) are contained within a database supplied with the code. For cases where the chemical is not in the database, Perry's Chemical Engineers' Handbook (Ref. 141) or an equivalent reference is used to modify the database to include the chemical.

The source term is then dispersed using the same ALOHA code. The EPHA performs consequence assessment using both 95% adverse and 50% average historical SRS meteorological data (Ref. 142). The ALOHA code contains both a Gaussian and heavy gas model. Within the code, a determination is made, based on the physical properties of the chemical and the meteorological conditions, as to which code is appropriate for the release being performed. This determination consists of calculating the "Richardson" number. The Richardson number can be calculated manually as follows (Ref. 143):

$$R_{i0} = \frac{g(\rho - \rho_a)V_o}{\rho_a U_{10}^3(d)} \quad (\text{Eq. 3.4-9})$$

where:

R_{i0}	=	Richardson number
g	=	acceleration due to gravity (9.81 m-sec^{-2})
ρ	=	density of substance released (kg-m^{-3})
ρ_a	=	ambient air density (kg-m^{-3})
V_o	=	volumetric release rate ($\text{m}^3\text{-sec}^{-1}$)
d	=	diameter of the release (m)
U_{10}	=	wind speed at a height of 10 meters (m-sec^{-1})

If the Richardson number is ≤ 0.003 , the release should be treated as a passive neutrally buoyant release (e.g., Gaussian model).

Chemicals in binary solutions (e.g., diluted acids) use the same assumptions for source term determination. The difference is that the rate at which the solution is concentrating or diluting needs to be determined before the physical properties can be determined and input into the ALOHA database. This methodology is detailed in WSRC-TR-96-0404 (Ref. 144) and is outlined as follows.

The representative spill size to be analyzed is determined, along with the analytical time steps for the concentration/dilution analysis. The amount of each constituent in the solution (e.g., acid and water) that evaporates over the time step is determined using a model typically referred to as the Kawamura and McKay model (Ref. 145). The Kawamura and McKay model is a manual calculation normally performed within a spreadsheet. The release rate, R_r , in $\text{kg}\cdot\text{sec}^{-1}$, of each constituent within the solution is calculated as follows:

$$R_r = (A)(K_M)\left(\frac{M_w P_v}{RT}\right) \quad (\text{Eq. 3.4-10})$$

where:

- A = area of the evaporating puddle (m^2)
- K_M = mass transfer coefficient ($\text{m}\cdot\text{sec}^{-1}$)
- M_w = molecular weight of substance ($\text{kg}\cdot\text{kmol}^{-1}$)
- P_v = vapor pressure (Pa)
- R = universal gas constant ($8314 \text{ J}\cdot\text{kmol}^{-1}\cdot\text{K}^{-1}$)
- T = ambient temperature ($^{\circ}\text{K}$)

The mass transfer coefficient, K_M , is calculated by the following equation:

$$K_M = 0.0048 \left(U^{7/9} \right) \left(Z^{-1/9} \right) \left(S_c^{-2/3} \right) \quad (\text{Eq. 3.4-11})$$

where:

- U = wind speed at a height of 10 meters ($\text{m}\cdot\text{sec}^{-1}$)
- Z = pool diameter in the wind direction (m)
- S_c = laminar Schmidt number

The laminar Schmidt number, Sc , is calculated by the following equation:

$$Sc = \frac{\nu}{D_M} \quad (\text{Eq. 3.4-12})$$

where:

$$\begin{aligned} \nu &= \text{kinematic viscosity of the air (m}^2\text{-sec}^{-1}\text{)} \\ D_M &= \text{molecular diffusivity of the chemical in air (m}^2\text{-sec}^{-1}\text{)} \end{aligned}$$

Graham's Law is used to approximate the molecular diffusivity, D_M , of the chemical as follows:

$$D_M = D_{H_2O} \left(\sqrt{\frac{M_{WH_2O}}{M_{WM}}} \right) \quad (\text{Eq. 3.4-13})$$

where:

$$\begin{aligned} D_{H_2O} &= \text{molecular diffusivity of water (2.4E-05m}^2\text{-sec}^{-1}\text{ @ 8 }^{\circ}\text{C)} \\ M_{WH_2O} &= \text{molecular weight of water (18 kg-kmol}^{-1}\text{)} \\ M_{WM} &= \text{molecular weight of the chemical (kg-kmol}^{-1}\text{)} \end{aligned}$$

The amount of each constituent evaporated over the analytical time step is subtracted from the original amounts, and the new concentration is calculated. The physical properties are adjusted in the Kawamura and McKay model, and the next step is run. For chemicals that are diluting (the vapor pressure is being reduced), the typical analysis uses relatively small time steps over a 15-minute period to calculate the average vapor pressure over the first 15 minutes of the release. For chemicals that are concentrating, the typical analysis starts with small time steps until a trend is developed. Once a trend is developed, the time step is adjusted until a time period of about 8 hours is reached. The physical properties at the end of the 8-hour period are used for the final source term determination.

If the final binary chemical solution source term is determined by a manual calculation with the consequence assessment being performed by direct input into the ALOHA code, the Richardson number is calculated using Equation 3.4-9 to determine whether to model the release as a neutrally buoyant gas (Gaussian) or a heavy gas.

Chemical Consequences Comparison to Guidelines

Calculated air concentrations are compared with the nonradiological risk acceptance criteria (accident frequency dependent) as prescribed in WSRC-MS-92-206 (Ref. 5). Appropriate toxicity limits (e.g., PEL-TWA, ERPG values, or ERPG-equivalent value) are based on those limits documented in S-CLC-G-00074 (Ref. 139).

The consequences are evaluated in sequence for the 100- and 640-meter onsite receptors and then for the 11.9-km offsite MOI receptor. If the onsite receptor concentration is shown to be far below the EG for the accident frequency class, no further evaluation is performed. It can be shown that more distant receptor locations will have much smaller concentrations using the same stability class and wind speed. For the distances to the boundary for the MOI calculation, the stable conditions of F Stability and 1.5 m/sec will lead to negligible concentrations relative to the proper EG.

The following table indicates the nonradiological risk guidelines used for the appropriate accident frequencies.

Nonradiological Risk Guidelines

Event Frequency (yr ⁻¹)	Primary Concentration Guidelines	
	Onsite	Offsite
<10 ⁻⁶	>ERPG-3	>ERPG-2
>10 ⁻⁶ to <10 ⁻⁴	ERPG-3	ERPG-2
>10 ⁻⁴ to <10 ⁻²	ERPG-2	ERPG-1
>10 ⁻² to <10 ⁻⁰	ERPG-1	PEL-TWA

3.4.1.4 RBOF Criticality Analyses

INTRODUCTION

The RBOF is designed to receive, store, and condition spent nuclear fuel, generally from offsite reactors. This facility has been in operation since 1963 and contains a large variety of spent nuclear fuel, in the form of fuel assemblies, bundles of fuel assemblies, and pieces of fuel assemblies. Spent fuel is stored in the RBOF under the authorization and specifications of NSDSs (Ref. 14), where the NSDSs are the documents that define the spacing and storage requirements of fuel assemblies and bundles.

One class of accidents to be analyzed is process error-induced criticality. The analysis focuses on frequencies of occurrence and consequences of such events. In general, process error events that may lead to a criticality accident have been identified in the hazards analysis and are summarized as follows (note that due to different handling procedures, process error events involving HFIR fuel are different from those listed here) (Ref. 146, 147):

- Floating fuel container
- Fuel storage infraction

- Critical configuration in fuel bundle
- Configuration of fissile debris on basin floor
- Fuel spills from fuel bucket during fuel movement
- Fuel spills from shipping cask during fuel loading/unloading
- More than one basket out of cask
- Fast leak in 70-ton cask adds positive reactivity
- Disassembled pieces fall in an unsafe geometry on the basin floor
- Fuel dropped back into cask
- Cask overloading

As stated previously, a large variety of nuclear fuels are stored within the RBOF. A recent study (Ref. 148) indicated that more than 30 fuel types are contained within the #1 storage basin. Analyzing all the events that may result in a criticality accident for all fuel types stored in the RBOF would be prohibitive. Instead, a prioritization plan that addresses the types of fuels to be analyzed for the process error-induced criticalities identified in the hazards analysis was developed (Ref. 149). The reactivity characteristics for process error configurations identified in ECS-SIM-960025 and S-CLC-H-00350 (Ref. 147, 150) for nine types of fuel were analyzed for this SAR (Ref. 151 through 165):

- Heavy Water Components Test Reactor (HWCTR)
- Massachusetts Institute of Technology Reactor (MIT)
- MURR
- Sterling Forest Oxide (SFO)
- High Flux Beam Reactor (HFBR)
- HFIR
- Elk River Reactor (ERR)
- SRE
- Dresden

These analyses modeled the reactivity effects of self-interactions of these fuels (HWCTR fuel stored with HWCTR fuel, rows of MIT fuel interacting with other rows of MIT fuel, etc.) and interactions among the bounding types of fuel. These fuels were selected to bound the RBOF's fuel based on the study of ECS-CAT-95-0107 and ECS-CAT-960011 (Ref. 149, 166). The selection criteria were based on reactivity of the fuel, RBOF's reconfiguration, and expected operational activity in the RBOF.

The purpose of these analyses was to evaluate the reactivity of potential interactions between and among the bounding fuel types in the RBOF against various process error configurations identified in the hazards analysis and to determine if a criticality event would occur.

COMPUTER CODE METHODOLOGY

A vast number of computer codes are used in the nuclear industry for reactivity (criticality) calculations. These codes differ in the geometry they can model (e.g., one-dimensional versus general geometry), the solution method (deterministic [e.g., discrete ordinates] versus probabilistic [e.g., Monte Carlo]), and the nuclear data libraries and neutron groups that they employ (two-, four-, sixteen-, ... , n -groups versus continuous energy).

In performing the criticality calculations documented in this report, the MCNP 4A code is used (Ref. 167). MCNP is a general-purpose Monte Carlo radiation transport code. MCNP numerically simulates neutral and charged particle transport problems and can be operated in any of the following modes: neutron transport only; photon transport only; coupled neutron/gamma transport; or coupled photon/electron transport. The MCNP code was developed by, and is maintained at, the Los Alamos National Laboratory.

The MCNP code has been extensively used to do the following:

- Analyze physics experiments and reactor designs
- Design nuclear safeguards nondestructive assay systems, nuclear shields, and nuclear instrumentation
- Calculate material activations
- Perform criticality analyses
- Address health physics problems

MCNP is also used for radiological dose determination, space craft radiation modeling, and radiation damage studies.

The MCNP code has been successfully used worldwide at hundreds of installations, and over 300 person-years of development and refinement have been invested in it. The intensive testing, modeling, and usage of the MCNP code make the possibility of significant errors in the code, or its employed cross-section data, very remote.

The Monte Carlo method, which is used in the MCNP code, is a numerical procedure based on probability theory. Monte Carlo methods are used to obtain answers to selected questions by simulating the history of individual particles. By following individual particles and recording their interactions, the average behavior of particles in a physical system is obtained from the average behavior of the individual particles tracked.

Advantages of the Monte Carlo method, as used by the MCNP code, include the ability to model complex geometries and the employment of extensive continuous-energy data libraries. A disadvantage of the Monte Carlo method is the extensive computer time required to obtain statistically meaningful information.

The code used for the RBOF analysis is versatile with many features:

- Employs a variety of nuclear data tables, which include the following:
 - Continuous energy and thinned continuous energy, discrete reaction, and multi-group reaction neutron interaction data
 - Neutron dosimetry cross-sections
 - Neutron $S(\alpha, \beta)$ data (to model thermal neutron scattering)
 - Neutron/photon interaction data
 - Photon interaction data
 - Coupled electron/photon interaction data
- Models general geometry and solves time-dependent transport problems
- Operates on many computer systems and environments (e.g., mainframes and UNIX-based workstations)
- Uses surfaces to create zones or regions. (The basic unit of MCNP geometry is the cell. Combinations of sense-signed surfaces are used to define regions of space encompassed by a cell in the ordinary Cartesian coordinate system. There are 26 standard surfaces [e.g., planes, spheres, cylinders, cones] defined in MCNP by first-, second-, and fourth-order equations that describe three-dimensional geometries.)
- Allows user input to describe a general source without any code modification. (Independent probability distributions may be specified for the source variables of energy, time, position, and direction. Information about the geometrical extent of the source can also be given.)
- Uses a variety of variance reduction techniques, which include the following:
 - Geometry splitting and Russian Roulette
 - Implicit capture and weight cutoff
 - Time and energy cutoffs
 - Forced collision
 - Weight windows
- Allows runs to be restarted
- Prints output in user-defined particle tallies (e.g., flux, current, heating) as a function of cell or surface, energy, time, angle, etc. (Tally modifiers allow easy tally

conversion to the quantities of interest, which include reaction rates such as absorption, fission, and fission heating.)

- Allows geometry plotting, using an interactive plotter to graph two-dimensional slices of the problem geometry specified in the input file
- Employs extensive error analysis procedures that give statistical error or uncertainty associated with the results
- Calculates k_{eff} and neutron removal lifetime. (The calculation is performed as a series of cycles or generations of neutrons in which both k_{eff} and removal lifetime are estimated in each cycle and are also averaged over all previous active cycles.)
- Employs a repeated structure capability, which allows the user to describe only once the cells and surfaces of any structure that appears more than once in a geometry. (This feature reduces the required amount of input data and computer memory. The amount of time required to set up a problem is also greatly reduced.)
- Supports two kinds of lattices: hexahedral and hexagonal prism lattices. (A lattice is created by the translation and/or rotation of a generic unit of an array to fill all space.)

For the RBOF criticality analyses, Version 4A of the MCNP code (configuration controlled on the IBM RISC/6000 workstation platform [Ref. 168, 169]) was employed in neutron-only transport mode to calculate nuclear eigenvalues for a variety of normal and accident configurations of stored fuel.

A variety of different materials were employed in these analyses. In every case, the default MCNP cross-section libraries were used (Ref. 167). In all cases but one, these libraries were derived from Version V of the Evaluated Nuclear Data File (ENDF/B-V) (Ref. 169). For modeling natural iron, data from recently completed evaluations of selected isotopes prepared by Group T-2 at Los Alamos National Laboratory were used (Ref. 167). In addition, MCNP $S(\alpha, \beta)$ tables were also employed, where appropriate, to model scattering of neutrons below 4 eV.

To account for the statistical uncertainty in the MCNP calculations, the calculated k_{eff} is adjusted as follows:

$$\text{adjusted } k_{\text{EFF}} = k_{\text{EFF}}^{\text{MCNP}} + 2\sigma_s \quad (\text{Eq. 3.4-14})$$

where σ_s is the statistical uncertainty (standard deviation) calculated by the MCNP code. The standard deviation is multiplied by a factor of two to achieve the 95.4% confidence level, assuming a normal distribution.

A configuration was considered critical if the final k_{eff} was at or above the acceptance criterion. Values for this acceptance criterion varied, depending on the fuel types being analyzed, and were calculated using the following equation:

$$k_{\text{CRITERION}} = 1.0 - \sigma_B - \sigma_M \quad (\text{Eq. 3.4-15})$$

where:

σ_B = MCNP code bias (which includes the effects of the bias uncertainty) for a particular fuel

σ_M = a margin ensuring subcriticality and accounting for any extension to the benchmark area of applicability for the analyzed fuel

In all cases, bounding biases and margins were employed.

Configurations in which the final k_{eff} was at or above this acceptance criterion were assigned a conditional probability for criticality of 1.0 in the fault trees. Any process error-induced postulated criticalities with final k_{eff} values below this criterion were assigned a conditional probability for criticality of 1.0E-06.

SCENARIOS MODELED

The following sections summarize the specific scenarios modeled to evaluate the reactivity of potential interactions between and among the bounding fuel types in the RBOF.

MURR

The MURR is an MTR whose core is a series of hollow, concentric cylinders of aluminum-clad, U-Al fuel (with the uranium enriched in U-235). Each MURR fuel assembly represents one-eighth of a MURR core and consists of 24 variously curved U-Al fuel plates clad with aluminum held between two aluminum side plates. MURR fuels are stored in aluminum general-purpose bundling tubes, with two fuel assemblies per bundle.

Six classes of interactions were modeled for the MURR fuel. In the first class of interactions, the safe number of MURR assemblies was calculated. In terms of the RBOF accident analysis, the safe number of any fissile unit is defined as the number of fissile units that can be arranged in any configuration in water without exceeding a specific, fuel-dependent reactivity acceptance criterion. In addition, the reactivity of this configuration, following any credible single failure or the addition of one more fissile unit, must also meet appropriate acceptance criteria to ensure subcriticality (Note that this form of the safe number is applicable to Chapter 3 analyses only and may not be equivalent to safe numbers documented in Nuclear Criticality Safety Evaluations or Chapter 6). As part of this analysis, groups of three, four, and five MURR assemblies were modeled in a variety of configurations (bounding configurations were chosen, without reference to any NSDS spacing limits).

In the second class of interactions, groups of two and six MURR fuel assemblies were modeled in transfer buckets. The transfer buckets are used to transfer MURR fuel assemblies from the cask basin to the repackaging basin. This set of analyses examined the reactivity effects of using the wrong transfer bucket inserts, thus allowing more than two MURR fuel assemblies in a

transfer bucket. Additionally, the reactivity effects of having more than one transfer bucket within 12 inches of each other were examined (each bucket was modeled to contain two MURR fuel assemblies).

The third class of MURR interactions involved the use of the Batelle Memorial Institute (BMI) cask basket. In this set of analyses, the reactivity effects of dropping two MURR fuel assemblies on top of, or adjacent to, a BMI basket fully loaded with MURR fuel were examined. The reactivity effects of having more than one BMI basket within 12 inches of each other were also examined.

In the fourth class of interactions, the safe number of MURR fuel bundles was calculated (as opposed to individual assemblies). In performing this set of analyses, groups of four, five, six, and seven bundles were modeled in various configurations.

In the fifth class of interactions, the reactivity effects of MURR fuel bundles in the water-filled, 70-ton shipping cask were analyzed. In the first phase of this analysis, the limiting 70-ton cask loading configuration for MURR bundles was calculated. Then, based on this limit, the reactivity effects of overloading the cask, misloading the cask, overloading individual bundles in the cask, and draining the cask (either partially or fully) were examined.

Finally, in the sixth class of modeled MURR interactions, various configurations of MURR fuel bundles in RBOF vertical row storage were analyzed. In this set of analyses, an infinite number of infinitely long rows of MURR bundles were modeled with a variety of spacing schemes. The examined storage configurations assumed a variety of violations of the MURR NSDS requirements. Examples of these modeled violations include the reduction of along-row and across-row spacing of fuel bundles, and the replacement of empty spacer regions (required to follow every four MURR bundles in a storage rack) with additional MURR fuel. For further conservatism, and to account for potential bundle overloading, all MURR fuel bundles were modeled to contain six MURR fuel assemblies rather than the two authorized by the MURR NSDS. Furthermore, the rows of MURR bundles were modeled to be tilted in their storage racks, at the worst observed degree of tilting, with the direction of the tilting preferentially chosen to maximize reactivity.

MIT

The MIT fuel assemblies are a form of MTR fuel, where each assembly consists of 15 flat, parallel U-Al fuel plates clad with aluminum (with the uranium enriched in U-235), held at a 30° angle to two aluminum side plates. MIT fuel can be stored in the RBOF as individual assemblies or in aluminum square-tube dissolver bundles, with three assemblies per bundle.

Six classes of interactions were modeled for the MIT fuel. In the first class of interactions, the safe number of MIT assemblies was calculated. As part of this analysis, groups of four, five, six, seven, eight, nine, and ten assemblies were modeled in a variety of configurations (bounding configurations were chosen, without reference to any NSDS spacing limits).

In the second class of interactions, MIT fuel assemblies were modeled in transfer buckets. This set of analyses examined the reactivity effects of using the wrong transfer bucket inserts, thus allowing more than three MIT fuel assemblies in a transfer bucket. Additionally, the reactivity effects of having more than one transfer bucket within 12 inches of each other were examined (with each bucket containing three MIT fuel assemblies).

The third class of MIT interactions involved the use of the BMI cask basket. In this set of analyses, the reactivity effects of dropping two MIT fuel assemblies on top of, or adjacent to, a BMI basket fully loaded with MIT fuel were examined. The reactivity effects of having more than one BMI basket within 12 inches of each other were also examined.

In the fourth class of interactions, the safe number of MIT fuel bundles was calculated. In performing this set of analyses, groups of three, four, five, six, seven, eight, nine, and ten bundles were modeled in various configurations.

In the fifth class of interactions, the reactivity effects of MIT fuel bundles in the water-filled, 70-ton shipping cask were analyzed. In the first phase of this analysis, the limiting 70-ton cask loading configuration for MIT bundles was calculated. Then, based on this limit, the reactivity effects of overloading the cask, misloading the cask, overloading individual bundles in the cask, and draining the cask (either partially or fully) were examined.

Finally, in the sixth class of modeled MIT interactions, various configurations of MIT fuel bundles in RBOF vertical row storage were analyzed. In this set of analyses, an infinite number of infinitely long rows of MIT bundles were modeled with a variety of spacing schemes. The examined storage configurations assumed a variety of violations of the MIT NSDS requirements. Examples of these modeled violations include the reduction of along-row and across-row spacing of fuel bundles, and the replacement of empty spacer regions (required to follow every three MIT bundles in a storage rack) with additional MIT fuel. For further conservatism, and to account for potential bundle overloading, all MIT fuel bundles were modeled to contain six MIT fuel assemblies rather than the three authorized by the MIT NSDS. Furthermore, the rows of MIT bundles were modeled to be tilted in their storage racks, at the worst observed degree of tilting, with the direction of the tilting preferentially chosen to maximize reactivity.

HWCTR

HWCTR driver assemblies are simple cylinders of a uranium-zirconium fuel (enriched in U-235), clad on the inside and outside surfaces with zirconium. HWCTR fuel is not stored in bundles or buckets, nor will it be transferred or unloaded from casks in the future. Therefore, all reactivity analyses of HWCTR fuel modeled individual assemblies, not bundles, baskets, or casks.

Two classes of interactions were modeled for the HWCTR assemblies. In the first class of interactions, the safe number of HWCTR assemblies was calculated. As part of this analysis, groups of four, five, seven, eight, nine, ten, eleven, and twelve assemblies were modeled in a

variety of configurations (bounding configurations were chosen, without reference to any NSDS spacing limits).

In the second class of modeled HWCTR interactions, various configurations of HWCTR assemblies in RBOF vertical row storage were analyzed. In this set of analyses, an infinite number of infinitely long rows of HWCTR assemblies were modeled with a variety of spacing schemes. The examined storage configurations assumed a variety of violations of the HWCTR NSDS requirements. Examples of these modeled violations include the reduction of along-row and across-row spacing of fuel bundles, and the placement of rows of HWCTR fuel adjacent to other rows of HWCTR fuel (storing HWCTR rows adjacent to HWCTR rows is not specifically allowed by the appropriate NSDS and thus was also modeled to be the product of a process error). In addition, the rows of HWCTR assemblies were modeled to be tilted in their storage racks, at the worst observed degree of tilting, with the direction of the tilting preferentially chosen to maximize reactivity.

SFO

SFO fuel consists of UO_2 powder (enriched in U-235), contained in aluminum cylindrical cans. The SFO cans may be stacked in aluminum square-tube dissolver bundles, with eight cans per bundle.

Four classes of interactions were modeled for the SFO fuel. In the first class of interactions, the safe number of SFO assemblies was calculated. As part of this analysis, groups of eight, nine, ten, and eleven cans were modeled in a variety of configurations (bounding configurations were chosen, without reference to any NSDS spacing limits).

In the second class of interactions, the safe number of SFO fuel bundles was calculated. In performing this set of analyses, groups of eight, nine, ten, eleven, and twelve bundles were modeled in various configurations.

In the third class of interactions, the reactivity effects of SFO fuel bundles in the water-filled, 70-ton shipping cask were analyzed. In the first phase of this analysis, the limiting 70-ton cask loading configuration for SFO bundles was calculated. Then, based on this limit, the reactivity effects of overloading the cask, misloading the cask, overloading individual bundles in the cask, and draining the cask (either partially or fully) were examined.

Finally, in the fourth class of modeled SFO interactions, various configurations of SFO fuel bundles in RBOF vertical row storage were analyzed. In this set of analyses, an infinite number of infinitely long rows of SFO bundles were modeled with a variety of spacing schemes. The examined storage configurations assumed a variety of violations of the SFO storage requirements, including the reduction of along-row and across-row spacing of fuel bundles. For further conservatism, and to account for potential bundle overloading, all SFO fuel bundles were modeled to contain ten SFO fuel cans rather than the eight cans calculated in the safe number analysis. Furthermore, the rows of SFO bundles were modeled to be tilted in their storage racks,

at the worst observed degree of tilting, with the direction of the tilting preferentially chosen to maximize reactivity.

HFBR

The HFBR is an MTR operated by Brookhaven National Laboratory. The bounding HFBR fuel is referred to as Type KM HFBR fuel. Each Type KM HFBR assembly consists of 18 curved aluminum-clad Al-U₃O₈ fuel plates (with the uranium enriched in U-235), two curved aluminum (nonfuel-bearing) plates, and two aluminum side plates. HFBR fuel can be stored in the RBOF as individual assemblies or in aluminum square-tube dissolver bundles.

Three classes of interactions were modeled for the HFBR fuel. Additional scenarios involving dropped fuel assemblies, etc., that were modeled for other fuel types were screened out as having insignificant reactivity effects in the HFBR analyses. In the first class of interactions, HFBR fuel assemblies were modeled in transfer buckets. This set of analyses examined the reactivity effects of using the wrong transfer bucket inserts, thus allowing more than five or six HFBR fuel assemblies in a transfer bucket. Additionally, the reactivity effects of having more than one transfer bucket within 12 inches of each other were examined (with each bucket containing five or six HFBR fuel assemblies).

The second class of HFBR interactions involved the use of the LWT seven-element cask basket. In this set of analyses, the reactivity effects of dropping two HFBR fuel assemblies on top of, or adjacent to, an LWT basket fully loaded with HFBR fuel were examined. The reactivity effects of having more than one LWT basket within 12 inches of each other were also examined.

Finally, in the third class of modeled HFBR interactions, various configurations of HFBR fuel bundles in RBOF vertical row storage were analyzed. In this set of analyses, an infinite number of infinitely long rows of HFBR bundles were modeled with a bounding spacing scheme. The examined storage configurations assumed a variety of violations of the HFBR NSDS requirements. Examples of these modeled violations include the reduction of along-row and across-row spacing of fuel bundles, and the replacement of empty spacer regions with additional HFBR fuel. For further conservatism, and to account for potential bundle overloading, all HFBR fuel bundles were modeled to contain the maximum number of HFBR fuel assemblies that could physically fit. Furthermore, the rows of HFBR bundles were modeled to be tilted in their storage racks, at the worst observed degree of tilting, with the direction of the tilting preferentially chosen to maximize reactivity.

In addition, the HFBR fuel was modeled according to the storage configuration associated with the newly installed Expanded Basin Storage (EBS) racks. Due to the inherent safe structural configuration, the only interaction of concern is a dropped fuel bundle on top of the EBS racks. This scenario was evaluated in N-NCS-H-00060 (Ref. 170).

ERR

The ERR fuel is stainless steel-clad rods of $\text{UO}_2\text{-ThO}_2$, with the uranium enriched in U-235. In the RBOF, ERR fuel rods are stored in aluminum general-purpose bundling tubes, with varying pitch and up to 128 rods per tube.

Two classes of interactions were modeled for the ERR fuel. Additional scenarios involving dropped fuel assemblies, etc., that were modeled for other fuel types were screened out as having insignificant reactivity effects in the ERR analyses. In addition, transfer bucket and cask basket analyses were not performed, because movement of this fuel type is not anticipated.

In the first class of interactions, the reactivity effects of the storage pitch of ERR rods in the bundling tubes were examined to provide bounding reactivities in all ERR analyses. The results of these analyses were then used in the subsequent class of interaction.

In the second class of modeled ERR interactions, various configurations of ERR fuel bundles in RBOF vertical row storage were analyzed. In this set of analyses, an infinite number of infinitely long rows of ERR bundles were modeled with bounding spacing schemes and bounding rod pitch. The examined storage configurations assumed a variety of violations of the ERR NSDS requirements. Examples of these modeled violations include the reduction of along-row and across-row spacing of fuel bundles, and the replacement of empty spacer regions with additional ERR fuel. Furthermore, the rows of ERR bundles were modeled to be tilted in their storage racks, at the worst observed degree of tilting, with the direction of the tilting preferentially chosen to maximize reactivity.

SRE

There are five distinct types of SRE fuel:

- Uranium metal slugs and hollow cylinders
- Uranium-molybdenum alloy slugs
- Uranium carbide slugs
- Uranium oxide pellets
- Thorium-uranium slugs

However, the current inventory of the RBOF indicates that only one uranium carbide can and 36 thorium-uranium cans are present in the RBOF. Therefore, these were the only SRE fuels analyzed. Both of these types of SRE fuel slugs are enriched in U-235. Both fuel types are stored in aluminum cans, with either five or seven vertical stacks of slugs per can (arranged with aluminum spacers).

Two classes of interactions were modeled for the SRE fuels. Additional scenarios involving dropped fuel assemblies, etc., that were modeled for other fuel types were screened out as having

insignificant reactivity effects in the SRE analyses. In addition, transfer bucket and cask basket analyses were not performed, because movement of this fuel type is not anticipated.

In the first class of interactions, the reactivity effects of various possible configurations of the SRE slugs in their storage cans were examined to provide bounding reactivities in all SRE analyses. The results of these analyses were then used in the subsequent class of interaction.

In the second class of modeled SRE interactions, various configurations of SRE fuel cans in RBOF vertical row storage were analyzed. In this set of analyses, an infinite number of infinitely long rows of SRE cans were modeled with bounding spacing schemes. The examined storage configurations assumed a variety of violations of the SRE NSDS requirements. Examples of these modeled violations include the reduction of along-row and across-row spacing of fuel cans, and the replacement of empty spacer regions with additional SRE fuel. Furthermore, the rows of SRE cans were modeled to be tilted in their storage racks, at the worst observed degree of tilting, with the direction of the tilting preferentially chosen to maximize reactivity.

Dresden

There are two distinct types of Dresden fuel: (1) low U-235 enriched UO_2 rods, and (2) highly enriched (in U-235) UO_2 - ThO_2 rods. Both rods employ stainless steel as the cladding material. The UO_2 - ThO_2 rods are contained in 23-tube, 34-tube, and single-tube encapsulation containers while the UO_2 rods are contained in three-tube encapsulation containers.

Two classes of interactions were modeled for each of the Dresden fuel types and for each of the possible storage containers. Additional scenarios involving dropped fuel assemblies, etc., that were modeled for other fuel types were screened out as having insignificant reactivity effects in the Dresden analyses. In addition, transfer bucket and cask basket analyses were not performed, because movement of this fuel type is not anticipated.

In the first class of interactions, the reactivity effects of various parametric changes to fuel container orientation, thickness, etc., were examined to provide bounding reactivities in all Dresden analyses. The results of these analyses were then used in the subsequent class of interaction.

In the second class of modeled Dresden interactions, various configurations of each Dresden fuel and container type in RBOF vertical row storage were analyzed. In this set of analyses, an infinite number of infinitely long rows of Dresden containers were modeled with bounding spacing schemes. The examined storage configurations assumed a variety of violations of the Dresden NSDS requirements. Examples of these modeled violations include the reduction of along-row and across-row spacing of fuel containers, and the replacement of empty spacer regions with additional Dresden fuel. Furthermore, the rows of Dresden cans were modeled to be tilted in their storage racks, at the worst observed degree of tilting, with the direction of the tilting preferentially chosen to maximize reactivity.

HFIR

The HFIR is a beryllium-reflected, light-water-cooled and -moderated MTR, operated by Oak Ridge National Laboratory. The core of the HFIR consists of two fuel elements: an inner annular element and an outer annular element. These cores both contain a large number of involute (curved) fuel plates held within cylindrical aluminum side plates. The fuel plates are composed of aluminum-clad Al-U₃O₈, with the uranium enriched in U-235. The inner fuel elements also contain a burnable B₄C neutron poison. HFIR fuel elements are stored individually in a special four-tiered storage rack that alternates inner and outer elements stored on stainless steel posts.

Because HFIR fuel elements are handled differently from the other fuel types considered in the SAR, a different group of process error-induced scenarios are considered (Ref. 147):

- Dropped HFIR cask
- HFIR fuel element brought too close to HFIR or other fuel in isolation
- Inner and outer HFIR elements brought too close in carrier stands in isolation
- HFIR fuel element dropped during transport to storage rack
- HFIR fuel element brought to rest on top of HFIR fuel in storage
- Inner HFIR fuel element lowered into a position between its proper post and a previously stored outer element
- HFIR fuel element dropped between previously stored HFIR fuel
- A section of the HFIR storage rack grating falls on top of a lower tier
- Outer HFIR fuel elements placed adjacently in storage

Detailed criticality analyses were modeled for each of the above scenarios that was determined to be credible. The examined configurations assumed a variety of violations of NSDS requirements. Examples of these modeled violations include the reduction of along-row and across-row spacing of fuel, and the replacement of empty spacer regions with additional fuel. Furthermore, rows of fuel adjacent to the HFIR storage rack were modeled to be tilted at the worst observed degree of tilting, with the direction of the tilting preferentially chosen to maximize reactivity.

CROSS-FUEL INTERACTIONS

The analyses described above modeled the effects of the bounding fuel types interacting only with fuel of their own type (Ref. 154, 155, 162). Three additional classes of interactions were analyzed to determine the bounding effects of cross-fuel interaction among a subset of the bounding fuel types (MURR, MIT, SFO, and HWCTR). The conclusions from these analyses were then extrapolated for the remaining bounding fuel types.

First, infinite numbers of infinitely long rows of bundles or assemblies were modeled, with a variety of storage violations assumed. Examples of these modeled violations include the reduction of along-row and across-row spacing of the fuel, the omission of empty spacer regions, and the overloading of bundles. In addition, the rows of fuel were modeled to be tilted in their storage racks, at the worst observed degree of tilting, with the direction of the tilting preferentially chosen to maximize reactivity. Using these row storage models, a variety of combinations of fuel were analyzed to bound any reactivity interaction effects among the fuel types for this class of cross-fuel interactions.

The second class of cross-fuel interactions modeled was identical to the first class, except that empty spacer regions were not omitted. Instead, the empty spacer regions were modeled to be erroneously filled with other types of fuel (MIT empty spacer regions were filled with either a single HWCTR assembly or with an SFO bundle; MURR empty spacer regions were filled with two HWCTR assemblies, one MIT bundle, or one SFO bundle, etc.). Using these modified row storage models (which include conservative storage violations similar to those assumed in the first class of cross-fuel interactions), a variety of combinations of fuel were analyzed to bound any reactivity interaction effects among the fuel types for this class of cross-fuel interactions.

Finally, in the third class of cross-fuel interactions, the reactivity effects of mispositioning or dropping fuel adjacent to rows containing various combinations of fuel types were analyzed. In these models, an infinite number of extremely long, but finite, rows of bundles or assemblies were modeled, with mispositioned fuel placed at regular intervals along the array. The modeled row storage configuration assumed a variety of storage violations, including the reduction of along-row and across-row spacing of fuel bundles, the omission of empty spacer regions, and the overloading of bundles. In addition, the rows of assemblies were modeled to be tilted in their storage racks, at the worst observed degree of tilting, with the direction of the tilting preferentially chosen to maximize reactivity. Using these models, diverse combinations of fuel storage rows were analyzed with a variety of mispositioned/dropped fuel (including self-interactions) to bound any reactivity interaction effects of dropped/mispositioned fuel.

HFIR fuel is stored on special hardware in one row of the RBOF. Accordingly, cross-fuel interactions between HFIR and ERR (the bounding fuel stored next to the HFIR row) were analyzed. Interactions between HFIR and HFBR assemblies in a transfer bucket were also analyzed to bound potential transfer accidents.

ASSUMPTIONS

In performing the analyses described above, a number of conservative assumptions were employed:

- All analyzed fuel elements were assumed to contain their unirradiated, fresh composition. The following effects of irradiation (burnup) of the fuel were conservatively neglected:

- Decreasing the amount of fissile material and thus decreasing reactivity
- Increasing the amount of neutron poisons via the production of fission products and thus decreasing reactivity
- The fuel plates of HFIR inner fuel elements each contain 0.1131 grams of B_4C burnable poison. Note that as an HFIR core is irradiated, the fresh fuel composition will decrease, decreasing reactivity; the fission product inventory will increase, decreasing reactivity; and this poison loading will decrease, increasing reactivity. However, time-dependent calculations of fuel, poison, fission product, and other isotopic concentrations indicate that the maximum reactivity of the HFIR fuel elements is at the beginning of the cycle, with fresh fuel and fresh poison. Therefore, for the HFIR reactivity analyses, B_4C poison is modeled in the inner fuel elements. However, for additional conservatism, the poison loading is decreased by 10% in the analyses (to 0.1018 grams per plate).
- Stainless steel storage posts in the HFIR storage rack act as neutron poison, reducing the reactivity of the stored fuel. To increase conservatism in the analyses, the minimum tolerance outer diameters are used (4.24 inches and 10.24 inches, for inner and outer element posts, respectively), and the post thickness are reduced by 20% (from 0.25 inches to 0.20 inches).
- The effects of the end parts (top and bottom) of most fuel types in row storage were ignored. That is, fissile regions of the fuel assemblies stored in bundles/containers were assumed to be in contact, and the vertical separation distance between stacked fuel elements was neglected.
- The bounding azimuthal orientation of the fuel was modeled.
- The SFO canisters normally contain UO_2 powder. Although water leaking into the SFO canisters is highly unlikely, and the solubility of the UO_2 in H_2O is less than optimum, the analyses assumed that water had leaked into the canisters and mixed with the UO_2 . The UO_2 and H_2O were assumed to be perfectly mixed, forming a homogeneous mixture of UO_2 in H_2O that fills the entire fuel region of the can. This conservative assumption bounds other possible fuel mixtures.
- In the configurations in which across-row reactivity effects were examined, all rows were either assumed to be infinitely long or at least as long as the actual row length, with an infinite number of rows modeled.
- Across-row tilting of fuel was preferentially modeled to produce the most reactive configuration.
- Fuel rods (Dresden and ERR) and slugs (SRE) in containers and cans were stored so as to maximize the reactivity of these fissile units.
- The widths of the cadmium poison plates in the 70-ton cask were conservatively set to the minimum thickness of 0.03 inches, as opposed to the nominal thickness of 0.04 inches.

- The actual models employed in the analysis assumed multiple process errors had occurred. Combinations of the following process errors (operator errors) were examined:
 - Along-row and across-row spacing was reduced between fuel units.
 - Overloaded fuel bundles were analyzed. That is, although the NSDSs (and other analyses) for the fuel types specify limited numbers of fuel assemblies per bundle, the analyses assumed additional assemblies were inadvertently loaded into the bundles.
 - Empty spacer regions were assumed to be filled with fuel identical to the type of fuel in that row or to be filled with other fuel types.
 - Rows of HWCTR assemblies were modeled adjacent to other rows containing HWCTR assemblies. The current NSDS does not permit adjacent-row storage of HWCTR assemblies.
 - Rows of MURR and MIT fuel were modeled adjacent to rows containing HWCTR assemblies. The current NSDSs do not permit MURR/MIT storage adjacent to HWCTR.
 - The wrong transfer bucket inserts are assumed to be used.
 - Multiple transfer buckets and cask baskets are assumed to be brought together, although procedures require a 12-inch separation.
- The effects of concrete reflection were not explicitly modeled in the majority of the process error-induced reactivity analyses. Instead, groups of fissile units were assumed to be surrounded on all sides by at least 30 cm of water (to approximate an infinite light-water reflector). For row storage of fuel elements, this is an accurate representation. However, for other scenarios involving small groups of fuel assemblies (i.e., safe number analyses), concrete may be present as a basin floor, wall and floor, or three-sided corner. To demonstrate that the effects of concrete reflection would not affect the conclusions of the safe number analyses (considered representative of the other analyses), additional calculations were performed with MCNP to model the reactivity effects of concrete reflection for MIT, MURR, HFBR, HWCTR, and HFIR fuel (Ref. 171, 172, 173). These calculations involved placing a safe number of each of these fuel types at varying proximities from one, two, and three concrete walls. The results of these calculations indicate that the reactivity effects of concrete are relatively small (~0% to 2% in the worst case) and do not affect the safety analyses.

In performing the analyses described above, a number of assumptions that limit operations (unless authorized by the performance of a USQE) in the RBOF and impose requirements were employed:

- Dummy bundles are to be placed between Saxton bundles stored along a row.

- HWCTR (driver assemblies), ERR, SRE, and Dresden fuel will not be transferred or unloaded from casks in the future.
- HWCTR, ERR, SRE, and Dresden fuel will not be placed in buckets or baskets. HWCTR assemblies will not be placed in containers, cans, or bundles.
- HWCTR assemblies will not be stored in rows adjacent to rows containing HWCTR assemblies.
- HFBR assemblies will be shipped to the RBOF in the seven-element baskets and LWT casks. MURR and MIT fuel will be shipped in the BMI basket. Other less reactive MTR fuel may be shipped with either configuration or different configurations provided the USQE authorizes such receipt.
- Future fuel shipments, in and out of the RBOF (without the use of the 70-ton shipping cask), are limited to HFIR fuel in the HFIR/GE cask, HFBR fuel or less reactive fuel in the LWT cask, and MURR and MIT fuel or other less reactive MTR fuel in the BMI cask.
- Future fuel shipments, in and out of the RBOF, using the 70-ton shipping cask are limited to SFO, MIT, and MURR fuel and other less reactive fuel. These types of fuel will only be shipped in the 70-ton cask as bundles, not in buckets or as individual assemblies or cans.
- Under normal conditions, 70-ton shipping casks transporting SFO, MURR, or MIT bundles will be filled with water.
- The limiting loading configuration for SFO bundles in the 70-ton shipping cask is nine or less bundles per slot, with slots 1, 3, and 5 filled. Other more reactive 70-ton shipping cask storage configurations are not allowed for SFO bundles.
- The limiting loading configuration for MIT bundles in the 70-ton shipping cask is eight or less bundles per slot, with slots 1, 3, and 5 filled (Ref. 174). Other more reactive 70-ton shipping cask storage configurations are not allowed for MIT bundles.
- The limiting loading configuration for MURR bundles in the 70-ton shipping cask is six or less bundles per slot, with slots 1 and 5 filled (Ref. 174). Other more reactive 70-ton shipping cask storage configurations are not allowed for MURR bundles.
- Type KM HFBR assemblies will be the only (or bounding) type of HFBR fuel accepted at the RBOF.
- HFBR assemblies will be stored in the RBOF in square-tube dissolver bundles in MTR-9T hardware in 9-inch rows with hardware set at beams 1 and 6.
- Either the five- or six-assembly transfer bucket insert (or any insert allowing less than five assemblies per bucket) will be used for all transfer bucket operations involving HFBR fuel.

- Other transfer bucket inserts allowing more than six assemblies per bucket, or allowing more optimum configurations of five or six assemblies per bucket, must not be made available for inadvertent use.

3.4.1.5 Criticality Event Fission Yield

Previous surveys have not identified criticality data applicable to spent fuel storage facility conditions (Ref. 76). Thus, a study was performed to review relevant metal and oxide system data from experiments and criticality incidents (Ref. 77). Distributions were then prepared based on fitting the applicable data (Ref. 175). From these distributions, estimates are made of the likelihood of exceeding a given fission yield level. Included in the Williamson study are three sets of data:

- Data Set A Nine accidents with bare- and reflected-uranium metal systems
- Data Set B Fifteen accidents with water-moderated metals and oxides
- Data Set C Data Set B plus Destructive test data sets (SL-1, BORAX, SPERT)

The "B" data contain 15 accidents with water-moderated metal and oxide assemblies in which a power spike, or burst, was observed. The boundary and initial conditions for the excursions among these data are similar to what would be expected in the RBOF during a process-induced criticality (e.g., one caused by mishandling or a procedural violation). Thus, in lieu of direct data for spent fuel storage criticalities, the "B" data constitute a useful surrogate from which a reference yield level may be determined.

A "C" data set is also defined by Williamson, formed by adding data from the SL-1, the BORAX, and the SPERT destructive tests to the 15-accident "B" data set. The three added destructive test cases are ones in which large sources of reactivity were inserted very rapidly into full reactors. In these experimental conditions, the fuel was mechanically restrained. In the subsequent analysis, the "C" data set shall be used for criticalities caused by external events (i.e., NPH-induced events that can be characterized as potentially large reactivity insertion events coupled with mechanical constraint boundary conditions on the spent fuel in the RBOF).

EXCEEDANCE PROBABILITY FOR A FISSION YIELD

Figure 3-4.1 shows the exceedance probability plot for both the "B" and "C" data sets developed by Morton for the Williamson analysis (Ref. 175). This plot indicates the probability that a numerical criticality yield will be exceeded, given a criticality incident has occurred and based on the two data set yields. For example, the median (50%) fission yield from one of the distributions is found by reading the value of the abscissa (x-axis) corresponding to intersection of the ordinate (y-axis) 0.5-intercept with the curves, or approximately $1.1\text{E}+17$ and $1.2\text{E}+17$ fissions for data sets "B" and "C," respectively. Similarly, the 95th-percentile fission yield, a level that would be expected to be exceeded 5% of the time based on the data set, is read from

the 0.05-intercept and is approximately $5\text{E}+17$ and $3.6\text{E}+18$ fissions, again for the "B" and "C" distributions, respectively.

BASIS FOR REFERENCE YIELD LEVEL

In this safety analysis, the exceedance probability curves derived (Ref. 175) from the data in EPD-CTG-950067 (Ref. 77) are used with the frequency estimate to derive a fission yield basis for the internal (process error-induced) class and two classes of external event criticality considered. The latter classes include those events induced by high winds and seismic events. The criticality estimates derived in this manner will be compared to earlier authorization basis fission yield levels.

Table 3.4-1 integrates frequency analysis for criticality events (discussed in Section 3.4.1.2) and postulated fission yield levels read from the exceedance probability distributions. Conditional probabilities and frequencies of exceeding various excursion sizes from $1.0\text{E}+17$ to $1.0\text{E}+20$ fissions are listed. The 95th-percentile fission yield is concluded to be sufficiently conservative for supporting safety accident analyses, and the 0.05 exceedance probability intercept is read from each distribution. Thus, from the "B" and "C" data sets, the fission yields of $4.6\text{E}+17$ fissions and $3.6\text{E}+18$ fissions are approximately equivalent evaluation basis levels for process error and high winds/seismic-induced events, respectively. These yield sizes encompass 95% of the exponential distribution obtained from a fit of the historically recorded events and experiments that are considered reasonably applicable to spent fuel under postulated criticality conditions in the RBOF. A margin of conservatism is applied, and the basis levels are rounded to $5.0\text{E}+17$ and $5.0\text{E}+18$, respectively.

Cliff effects, such as prompt excursions, certainly can be identified as criticality yields exceed $\sim 1.0\text{E}+20$. But the review of the historical data and the statistical fits for spent fuel systems as contained in the RBOF imply that this is not justifiable in design basis analysis.

The statistical fits of the available experimental data and accident assessments provide a rationale for evaluating whether higher yield criticalities should be treated in the context of a design basis analysis. Considering the class of process-induced criticality, Figure 3.4-1 and Table 3.4-1 indicate that beyond $5\text{E}+17$ fissions, there is a vanishingly small likelihood that higher yield events could be anticipated. For example, at $1\text{E}+18$ fissions, the exceedance frequency is $\sim 1\text{E}-07/\text{yr}$, and at $5\text{E}+18$ fissions, the exceedance frequency is below $1\text{E}-15/\text{yr}$. The data and internal events frequency then seem to argue that although high fission yield levels could be used, the analysis would then be clearly in beyond design basis space ($< 1\text{E}-06/\text{yr}$). Similarly, the external events can be viewed beyond $5\text{E}+18$ fissions, but again such levels are inappropriate for design basis consideration.

EVALUATION BASIS CRITICALITY YIELDS

The remainder of the safety analysis will use an excursion yield level discussed above for process error-induced criticality at $5.0\text{E}+17$ fissions, approximately the 95th percentile value of the "B" distribution. Both high wind and seismic classes of external events will use an evaluation basis level of $5.0\text{E}+18$ fissions, or a level that encompasses nearly 97% of the distribution obtained from a log-normal fit of the observed events contained in the "C" data set. It is observed from Table 3.4-1 that a level of $5.0\text{E}+18$ fissions has an exceedance probability of about 0.035 from the "C" data set. Thus, it would be expected that a postulated criticality event would exceed this yield value only once in about 29 excursions ($1/0.035$).

Selection of fission yield reference values from the available experimental and incident data is consistent with the arguments presented by British criticality safety analysts in selecting a yield level based on a criticality evaluation basis that encompasses about 90% of the excursions (Ref. 176).

SAR ADDENDUM AND BIO BASES

Previously, before the identification and application of these data collections to the safety analyses for the RBOF, earlier authorization basis documentation assumed slightly different estimates. The October 1993 SAR Addendum (Ref. 45) used a "mean value for solid systems" of $5.0\text{E}+17$ fissions along with a "bounding value ($1.0\text{E}+19$ fissions) for the maximum release estimate." It was further reasoned that the bounding value was consistent with NRC regulations. It may be noted that the Regulatory Guide referenced (Regulatory Guide 3.34) provides prescriptive assumptions for a uranium fuel fabrication plant, rather than a spent fuel storage facility containing mostly solid metal assemblies.

The BIO (Ref. 46) again used $5.0\text{E}+17$ fissions for evaluating the facility worker doses from internal event-caused criticality. The postulated burst of $5.0\text{E}+17$ fissions led to facility receptor doses of approximately 18 rem from inhalation and cloudshine due to fission gases (18 rem), and a direct radiation dose (12 rem), summing to a total dose of 30 rem. In this analysis, it was assumed that no basin water displacement occurred between the fission locus and the facility receptor. Facility worker doses were not calculated for NPH events. Receptor doses at 640 meters and the SRS boundary were evaluated using the bounding value of $1.0\text{E}+19$ fissions.

DOE SOURCE TERM HANDBOOK

DOE-HDBK-3010-94 (Ref. 75) discusses four categories of inadvertent nuclear criticality:

- Solutions
- Fully moderated/reflected solids
- Bare, dry solids

- Large storage arrays

Fission yield recommendations for the second (fully moderated/reflected solids) and the fourth (large storage arrays) categories are analogous to the process upset and external event criticality types assumed in the RBOF analysis, respectively.

Fully moderated/reflected solids include configurations covering "reflected bulk metal and metal pieces ... that are moderated or reflected." Mishima continues for this category

"Given the types of situations encountered in DOE facilities where it is difficult to accumulate the quantity of materials required, contain the material and moderator, and assume any shape that would be unfavorable, a reference value of $1\text{E}+18$ fissions in a single burst is assessed to be the bounding reference value and is believed to be very conservative."

As Table 3.4-1 indicates, adoption of the $1.0\text{E}+18$ fissions level for the RBOF analysis would implicitly accept a beyond design basis event category frequency. Moreover, the selection of $5.0\text{E}+17$ fissions from the Williamson/Morton review of applicable data as the reference basis for a process-induced criticality is comparable to the $1.0\text{E}+18$ fissions reference value recommended by Mishima in view of measurement and calculational uncertainties in heterogeneous, critical geometries. The reference yield of $5.0\text{E}+17$ fissions is concluded to be reasonable for SAR evaluation purposes.

3.4.1.6 Conservatism in Safety Analysis

In calculating consequences in safety analysis, a number of phenomenological uncertainties usually surface. While it is not appropriate to ignore these and only provide best estimates, it is also deemed not very appropriate and useful to compound them to arrive at ultraconservative results that have no real meaning. Instead, the following approach of considering the individual uncertainties and then combining them into an overall uncertainty factor to be applied to the best-estimate result is recommended as described in S-CLC-H-00426 (Ref. 177).

The target suggested for the overall amount of conservatism is that it bound 95% of the uncertainty distribution. With an accident initiator located centrally within a frequency bin of two orders of magnitude in width, it can be seen that any larger conservatism would give an overall frequency corresponding to the adjacent lower frequency bin.

The following discussion is only rigorously correct when (1) the consequence is a product of independent factors; (2) the factor uncertainties can be approximated by the lognormal distribution; and (3) the uncertainty in these factors is uncorrelated. Nevertheless, these criteria hold for many problems and are approximately correct or even conservative for many others. If dependencies or correlations are identified, there are software tools such as @RISK and CRYSTAL BALL that can be used to propagate the uncertainty.

Procedure

For each important factor (i.e., factors with uncertainties that can cause a major change in the overall consequence), the median (X-50% = value with 50% chance of exceedance) and the upper 95% error factor are estimated using the following equation:

$$EF_i = (x_{-95\%}/x_{-50\%}) \quad (\text{Eq. 3.4-16})$$

For the overall product of the factors, the overall median (equals product of the medians) and the overall 95% error factor are estimated using the following equation:

$$(\ln(EF))^2 = \sum (\ln(EF_i))^2 \quad (\text{Eq. 3.4-17})$$

The 95% upper bound is determined as follows:

$$\text{Overall median} * \text{Overall error factor} \quad (\text{Eq. 3.4-18})$$

Extension

In many dose applications, a single isotope and single release mechanism dominate the dose of interest. For these cases, the resultant consequence is effectively the simple product of factors for which the impact of an error factor for any factor translates into an impact of the same magnitude on the dose, and for which the methodology above directly applies. In other cases, there may be a combination of several release mechanisms or isotopes such that the dose is of the following form:

$$d = [(x_1 * x_2) + (x_3 * x_4)] * x_5 * x_6 \quad (\text{Eq. 3.4-19})$$

where a perturbation of some factors (those in parentheses) does not translate to the exact same impact magnitude on the dose. Even if one or the other of the terms in parentheses is not clearly dominant, the above approach should still provide reasonable results. The error factors for the terms in parentheses are replaced by a pseudo factor. A pseudo factor is determined by the factor of change in total dose as an individual factor (within parentheses) is varied from its median to upper bound (i.e., median*EF) value. An example that shows how a pseudo factor is determined is provided in Section 3.4.2.4.

Methodology Limitations

The use of the 95% bound is appropriate when looking at the spread range of a consequence for a fixed scenario type in comparing against the consequence limit for that scenario's frequency bin. In some instances, for example where there is an explosion potential that could fail the confinement, phenomena beyond the 95% bound may have to be addressed because the whole course of the accident could be changed. Initiators with a frequency at or above 1E-04/yr may well have to consider phenomena with conditional probabilities as low as 1% (or even lower) to

be sure unacceptably high consequences are not credible. In these kinds of situations where a cliff or discontinuity in consequence is credible, it is best to address both types of scenarios (below and above cliff) with their appropriate conditional probabilities, and the method proposed herein can be applied in determining the dose for both scenarios.

For the RBOF criticality accident analysis (Section 3.4.2.4), the above methodology has been applied to demonstrate the unrealistic doses that result when the uncertainty for all factors is combined simultaneously.

3.4.2 DESIGN BASIS ACCIDENTS

3.4.2.1 Chemical Spills

Three chemical spill events were identified in the hazards analysis as requiring unique accident analyses. Each of these three events is analyzed in this section. They include chemical spills of sodium hydroxide, nitric acid, and sodium nitrite from storage dumpsters located outside the facility. It should be noted that other dispersions (e.g., splashing) are not discussed in the three chemical spill events. It has been shown that the source term contribution from the splashing is at least an order of magnitude lower than the current analysis. In other words, the splashing source term is more than accounted for in the conservatism employed in the evaporative source term model (Ref. 178).

LOSS OF 50% SODIUM HYDROXIDE CONTAINMENT

Loss of sodium hydroxide containment will result in a chemical spill outside the facility. The hazards analysis categorized the risk of this event (Event 316) as having moderate consequences and an anticipated likelihood of occurrence, or a risk rank of 4.

Scenario Development

Sodium hydroxide (NaOH), used for anion resin regeneration, is stored in a skid-mounted, 1,000-gallon dumpster (EP52) on chemical pad No. 1, located to the west of Building 245-H. This dumpster is typically filled to 900 gallons and is heated during the winter months to approximately 10°F to 12°F above the crystallization point using 25-psi steam routed via piping inside the dumpster. Sodium hydroxide is supplied to the RRF by means of overhead piping. Flow to the RRF is controlled by two valves, OH-22 and OH-21. Two pumps located inside the RRF allow for a maximum flow of 118 gal/hr of material into the facility. Chemical pad No. 1 is located approximately at ground level outside the RBOF and RRF and is surrounded by a dike.

The dumpster is shipped to another facility in F Area to be refilled with the appropriate chemicals. Preparation for shipping involves the disconnection of associated piping, flexible

hoses, and instrumentation. Facility personnel follow the latest facility normal operating procedures while performing these operations.

Based on the hazards analysis, this accident analysis examines the frequency and consequences of a spill of 50% sodium hydroxide from dumpster EP52 due to causes such as material failure, mechanical damage, equipment malfunction, and human error. Sodium hydroxide spills due to earthquake or high winds, severe weather, and surface vehicle impacts are covered under those respective accident analysis sections.

Chemical spills associated with filling (e.g., overfilling) of the dumpster outside of the RBOF and RRF are not considered here.

Possible Initiating Events

The following are possible initiating events for this scenario:

- Material failure of the dumpster or associated piping, valves, flex hoses, or steam line occurs due to fatigue or corrosion.
- Dumpster outlet valve (OH-22) fails to close due to mechanical failures.
- The dumpster is dropped during movement due to human error or failure of the lifting mechanism.
- Dumpster outlet valve (OH-22) fails to close due to human errors.

Accident Event Progression

If an initiating event occurs, the chemical would accumulate in the bermed area of the pad. The rate of leakage from the dumpster will depend on the point of breach and the severity of the opening. A worst-case breach would drain the entire contents of the dumpster within minutes.

For this scenario, no credit is given for the berm containing the chemical spill. Therefore, the sodium hydroxide spreads over a large area, which results in a larger ARR than for a bermed release.

Preventive Features

A preventive feature for this scenario is that the dumpster outlet valve (OH-22) is normally shut.

Mitigative Features

The following are mitigative features for this scenario:

- The chemical storage dumpster is located in a bermed area that is sloped to waste tank EP38. The waste tank volume is 6,400 gallons.
- Sodium hydroxide has an attraction to water vapor. Therefore, the sodium hydroxide will never completely dry out, but rather will form a firm, moist pack of sodium hydroxide.

Frequency Binning

A fault tree was developed to quantify the frequency of spilling sodium hydroxide due to the initiating events listed above. The fault tree analysis is documented in S-CLC-H-00281 (Ref. 179). The frequency of the event, as estimated by the fault tree, is $5.28\text{E-}03/\text{yr}$, which is in the unlikely ($10^{-2}/\text{yr} > p > 10^{-4}/\text{yr}$) frequency bin.

The fault tree was developed and quantified using the CAFTA computer code (Ref. 180). Input data for each basic event, consisting of component failure rates and human error probabilities, are obtained from WSRC-TR-93-581, WSRC-TR-93-262, DP-1633, and NUREG/CR-1278-F (Ref. 60, 120, 181, 182). The logic of the fault trees is arranged into four major categories: material failure, equipment malfunction, mechanical damage, and human error.

The following assumptions were used as input and are specific to the fault tree evaluation performed for a sodium hydroxide spill:

- The facility is operated 24 hours a day, 365 days a year. Dumpster EP52 is always in operation, except when removed from the system for refilling.
- Dumpster EP52 always contains at least 72 gallons of sodium hydroxide. A large chemical spill from dumpster EP52 is defined in the RBOF hazards analysis as a spill greater than 72 gallons of 50% sodium hydroxide. A spill greater than 72 gallons will result in exceeding ERPG limits for a hazardous chemical release to the environment. **(Credit assumed for Hazardous Material Inventory Program)**
- Dumpster EP52 is stored on chemical pad No. 1. The top surface of this concrete pad is installed essentially at ground level. A dike is installed around the perimeter of the chemical pad to contain any leaks from the dumpster and associated piping systems. Success or failure of this dike to collect or contain the spill is not considered when determining the frequency of a sodium hydroxide spill.
- Chemical pad No. 1 contains a drain line to direct any chemical spill to waste tank EP38. No credit is taken for this drain line (i.e., assume drain line is clogged) when determining the frequency of a sodium hydroxide spill.
- Dumpster EP52 is typically filled approximately 15 to 18 times a year. The maximum number of refills anticipated for the dumpster is 24 times a year.
- No credit is taken for any inspection or testing performed on the truck-mounted uni-arm swing-out lever used for lifting the dumpsters from the chemical pads.

- A dumpster (or attached piping or valve) is breached when the dumpster is dropped during a loading/unloading operation. No credit is taken for any vessel certification to DOT standards or drop-test qualifications.
- No credit is taken for independent checking performed on proceduralized activities involving connection and disconnection of the dumpsters on the chemical pads.
- The following piping/hose lengths are assumed for this analysis:
 - A cumulative total of 15 feet of piping is connected between the sodium hydroxide dumpster and the metering pump inlet valve.
 - A total of 5 feet of flexible hose is connected between the dumpster and the respective metering pump inlet valve.
 - A total of 15 feet of steam piping is installed inside the sodium hydroxide dumpster.

These sections of pipe/hose are conservatively assumed to be constantly exposed to fluid and, therefore, always vulnerable to a large spill. Other "downstream" piping or valves are not considered since they are vulnerable only when transfers are in progress, which is only a small fraction of a year. Therefore, failures of these components are considered insignificant contributors to the overall spill frequency.

- Failure due to structural fatigue is not considered because the dumpsters and associated piping and valves are not subjected to cyclic structural loading in excess of the fatigue limit (i.e., 2 to 10 million cycles) for most steels (Ref. 183). Although corrosion due to caustic and acidic chemicals can substantially reduce fatigue strength, the low frequency of valve manipulations involving connection and disconnection of hoses and vessel movements associated with refilling activities (i.e., once per week at worst case) does not warrant analysis for corrosion fatigue. The dumpsters are vented to the atmosphere (i.e., nonpressurized vessels).
- Failure due to thermal stress is not considered because the dumpsters and associated piping and valves are not normally subjected to cyclic high temperature transients. The equipment is exposed to ambient conditions.

The dominant scenario (i.e., the cutset contributing the most to a potential sodium hydroxide spill) is due to a dropped dumpster resulting in mechanical damage leading to dumpster breach.

Source Term Analysis

The source term for the accident involving a loss of sodium hydroxide containment is a function of the total amount available inside the storage dumpster (MAR), the amount released from the dumpster (DR), the amount that drains into waste tank EP38, and the rate of airborne release. MAR and DR are terms typically used in determining the source term for radionuclide source terms, but they have applicability in the case of a chemical spill. The generalized equation for determining the source term that is evaporated and available for dispersion is as follows:

$$\text{Source Term} = \text{MAR} * \text{DR} * [1 - (\% \text{ drained})] * \text{ARR} * \text{duration} \quad (\text{Eq. 3.4-20})$$

where:

ARR = Airborne Release Rate

duration = total length of time used to determine the amount of material released

Because sodium hydroxide is a salt, as the water content of sodium hydroxide evaporates, most of the chemical will accumulate on the pad. Only a small fraction, based on the ARR, will be dispersed as part of the evaporating water. The source term calculation is documented in S-CLC-H-00278 (Ref. 184).

Material at Risk

The dumpster has a capacity of 1,000 gallons of sodium hydroxide (Ref. 13). Accounting for a possible 10% head space in the dumpster that could be filled, the MAR equals 1,100 gallons of 50% sodium hydroxide.

Damage Ratio

The DR is the fraction of the MAR that is impacted by the accident-generated conditions. For this scenario, it translates into the amount of material that would be released from the dumpster in the event of a breach or leak. This scenario assumes a total release of material. The DR is conservatively estimated to be 1.0.

Amount Drained to Waste Tank EP38

The dumpster is located on a bermed pad, which drains to waste tank EP38. The waste tank capacity is 6,400 gallons and overflows to the waste cell if exceeded. Requirements call for draining the waste tank when 75% full, which allows for a minimum available capacity of

$$6,400 - (0.75 * 6,400) = 1,600 \text{ gallons.}$$

Since the sodium hydroxide dumpster can never have more than 1,100 gallons of material, the waste tank can accommodate the entire contents of the sodium hydroxide dumpster. However, to ensure a conservative estimate for the source term calculation, it is assumed that no material drains to the waste tank. Therefore, all sodium hydroxide will remain within the surface surrounding the tank.

Airborne Release Rate

The sodium hydroxide will accumulate within the bermed area of chemical pad No. 1 and evaporate. Since sodium hydroxide is a salt solution, the water content of the chemical will evaporate while the salt component will accumulate and concentrate in the bermed area. A small fraction of the sodium hydroxide particulate will disperse with the evaporating water. This fraction, the ARR, is provided in DOE-STD-3010-94 (Ref. 75). Its value is

$$ARR = 4E-06/\text{hr.}$$

Total Source Term

Based on the discussion above, the rate of release of sodium hydroxide can be determined as follows:

$$\begin{aligned} \text{Source Term} &= 1,100 \text{ gallons} * \text{DR} * \text{ARR} * \text{duration} && (\text{Eq. 3.4-21}) \\ &= 1,100 * 1.0 * 4E-06/\text{hr} \\ &= 4.4E-03 \text{ gallons of 50\% NaOH per hour} \\ &= 4.4E-03 \text{ gal/hr} * (1 \text{ hr}/3600 \text{ sec}) * (6.364 \text{ lb. NaOH/gal}) * (1 \text{ kg}/2.205 \text{ lbs.}) \\ &= 3.5E-06 \text{ kg/sec of NaOH disbursed to the air.} \end{aligned}$$

Consequence Analysis

The sodium hydroxide plume will be passively transported by the wind as tiny particles. Therefore, a Gaussian plume model is appropriate. The Gaussian plume equation for neutrally buoyant gases is as follows (Ref. 185):

$$C = \frac{Q}{\pi \sigma_z \sigma_y u} \quad (\text{Eq. 3.4-22})$$

where:

- C = centerline, groundlevel concentration for a groundlevel release (kg/m³)
- Q = continuous source strength (kg/sec)
- u = wind speed (m/sec)
- σ_y = lateral dispersion parameter (m)
- σ_z = vertical dispersion parameter (m)

The dispersion parameters σ_y and σ_z are also included in Reference 185. They are dependent on the wind stability class and the downwind distance, x (meters).

Briggs-Rural Dispersion Parameters

Stability Class	σ_y (m)	σ_z (m)
A	$0.22x*(1+0.0001x)^{-1/2}$	$0.20x$
B	$0.16x*(1+0.0001x)^{-1/2}$	$0.12x$
C	$0.11x*(1+0.0001x)^{-1/2}$	$0.08x*(1+0.0002x)^{-1/2}$
D	$0.08x*(1+0.0001x)^{-1/2}$	$0.06x*(1+0.0015x)^{-1/2}$
E	$0.06x*(1+0.0001x)^{-1/2}$	$0.03x*(1+0.0003x)^{-1}$
F	$0.04x*(1+0.0001x)^{-1/2}$	$0.016x*(1+0.0003x)^{-1}$

The continuous source strength, Q , is $3.5E-06$ kg/sec. The wind speed, u , is assumed to be 4.5 m/sec for the onsite consequences. The lateral and vertical dispersion parameters, σ_y and σ_z , use a D stability class and a downwind distance of 100 meters. The parameters are therefore 8.0 meters and 5.6 meters, respectively.

Incorporating the appropriate values into the Gaussian plume equation, the concentration is $6E-03$ mg/m³.

Comparison to Guidelines

The lowest concentration guideline (PEL-TWA) for either onsite or offsite and independent of event frequency for sodium hydroxide is 2 mg/m³ (Ref. 5). For the sodium hydroxide analysis (Ref. 186), the concentration at the closest receptor was evaluated to be far below the most restrictive of the EGs (PEL-TWA). Other more distant receptor locations can be evaluated, but they would be found to be at substantially smaller concentrations. Therefore, the consequence quantification is stopped after the 100-meter location.

Summary of Safety-Class SSCs and TSR Controls

This accident scenario does not challenge the offsite EGs; therefore, no safety-class SSCs are required.

LOSS OF 50% NITRIC ACID CONTAINMENT

Loss of nitric acid containment will result in a chemical spill outside the facility. The hazards analysis categorized the risk of event 318 as having moderate consequences and an anticipated likelihood of occurrence, or a risk rank of 4.

Scenario Development

Nitric acid (HNO_3), used for cation resin regeneration, is stored in a skid-mounted, 1,000-gallon dumpster (EP50) on chemical pad No. 1, located to the west of Building 245-H. This dumpster is typically filled to 900 gallons. Nitric acid is supplied to the RRF by means of underground piping. Flow to the RRF is controlled by two valves, H-22 and H-21. Two pumps located inside the RRF allow for a maximum flow of 108 gal/hr of material into the facility. Chemical pad No 1 is located approximately at ground level outside the RBOF and RRF and is surrounded by a dike.

The dumpster is shipped to another facility to be refilled with the appropriate chemicals. Preparation for shipping involves the disconnection of associated piping, flexible hoses, and instrumentation. Facility personnel follow the latest facility normal operating procedures while performing these operations.

Based on the hazards analysis, this accident analysis examines the frequency and consequences of a spill of greater than 71 gallons of nitric acid from dumpster EP50 due to causes such as material failure, mechanical damage, equipment malfunction, and human error. Nitric acid spills due to earthquake or high winds, severe weather, and surface vehicle impacts are covered under those respective accident analysis sections.

Chemical spills associated with filling (e.g., overfilling) of the dumpster outside of the RBOF and RRF are not considered here.

Possible Initiating Events

The following are possible initiating events for this scenario:

- Material failure of the dumpster or associated piping, valves, or flex hoses occurs due to fatigue or corrosion.
- Dumpster outlet valve (H-22) fails to close due to mechanical failures.
- The dumpster is dropped during movement due to human error or failure of the lifting mechanism.
- Dumpster outlet valve (H-22) fails to close due to human errors.

Accident Event Progression

If an initiating event occurs, the chemical would accumulate in the bermed area of the pad. The rate of leakage from the dumpster will depend on the point of breach and the severity of the opening. A worst-case breach would drain the entire contents of the dumpster within minutes.

For this scenario, no credit is given for the berm containing the chemical spill. Therefore, the nitric acid spreads over a large area, which results in a larger ARR than for a bermed release.

Preventive Features

A preventive feature for this scenario is that the dumpster outlet valve (H-22) is normally shut.

Mitigative Features

The following are mitigative features for this scenario:

- The chemical storage dumpster is located in a bermed area that is sloped to waste tank EP38. The waste tank volume is 6,400 gallons.
- The facility normal operating procedures specify that the chemical pad drainage system be operable.

Frequency Binning

A fault tree was developed to quantify the frequency of spilling nitric acid due to the initiating events listed above. The fault tree analysis is documented in S-CLC-H-00281 (Ref. 179). The frequency of the event, as estimated by the fault tree, is $5.27\text{E-}03/\text{yr}$, which is in the unlikely ($10^{-2}/\text{yr} > p > 10^{-4}/\text{yr}$) frequency bin.

The fault tree was developed and quantified using the CAFTA computer code (Ref. 180). Input data for each basic event, consisting of component failure rates and human error probabilities, are obtained from WSRC-TR-93-581, WSRC-TR-93-262, DP-1633, and NUREG/CR-1278-F (Ref. 60, 120, 181, 182). The logic of the fault trees is arranged into four major categories: material failure, equipment malfunction, mechanical damage, and human error.

The following assumptions were used as input and are specific to the fault tree evaluation performed for a nitric acid spill:

- The facility is operated 24 hours a day, 365 days a year. Dumpster EP50 is always in operation, except when removed from the system for refilling.
- Dumpster EP50 always contains at least 71 gallons of nitric acid. A large chemical spill from dumpster EP50 is defined in the RBOF hazards analysis as a spill greater

than 71 gallons of nitric acid. A spill greater than 71 gallons will result in exceeding ERPG limits for a hazardous chemical release to the environment. **(Credit assumed for Hazardous Material Inventory Program)**

- Dumpster EP50 is stored on chemical pad No. 1. The top surface of this concrete pad is installed essentially at ground level. A dike is installed around the perimeter of the chemical pad to contain any leaks from the dumpster and associated piping systems. Success or failure of this dike to collect or contain the spill is not considered when determining the frequency of a nitric acid spill.
- Chemical pad No. 1 contains a drain line to direct any chemical spill to waste tank EP38. No credit is taken for this drain line (i.e., assume drain line is clogged) when determining the frequency of a nitric acid spill.
- Dumpster EP50 is typically filled approximately 15 to 18 times a year. The maximum number of refills anticipated for the dumpster is 24 times a year.
- No credit is taken for any inspection or testing performed on the truck-mounted uni-arm swing-out lever used for lifting the dumpsters from the chemical pads.
- A dumpster (or attached piping or valve) is breached when the dumpster is dropped during a loading/unloading operation. No credit is taken for any vessel certification to DOT standards or drop-test qualifications.
- No credit is taken for independent checking performed on proceduralized activities involving connection and disconnection of the dumpsters on the chemical pads.
- The following piping/hose lengths are assumed for this analysis:
 - A cumulative total of 10 feet of piping is connected between the nitric acid dumpster and the metering pump inlet valve.
 - A total of 5 feet of flexible hose is connected between the dumpster and the respective metering pump inlet valve.

These sections of pipe/hose are conservatively assumed to be constantly exposed to fluid and, therefore, always vulnerable to a large spill. Other "downstream" piping or valves are not considered since they are vulnerable only when transfers are in progress, which is only a small fraction of a year. Therefore, failures of these components are considered insignificant contributors to the overall spill frequency.

- Failure due to structural fatigue is not considered because the dumpsters and associated piping and valves are not subjected to cyclic structural loading in excess of the fatigue limit (i.e., 2 to 10 million cycles) (Ref. 183) for most steels. Although corrosion due to caustic and acidic chemicals can substantially reduce fatigue strength, the low frequency of valve manipulations involving connection and disconnection of hoses and vessel movements associated with refilling activities (i.e., once per week at worst case) does not warrant analysis for corrosion fatigue. The dumpsters are vented to the atmosphere (i.e., nonpressurized vessels).

- Failure due to thermal stress is not considered because the dumpsters and associated piping and valves are not normally subjected to cyclic high temperature transients. The equipment is exposed to ambient conditions.

The dominant scenario for a potential nitric acid spill is due to a dropped dumpster resulting in mechanical damage breaching the container.

Source Term Analysis

The source term calculated herein is for an unbermed spill of nitric acid. The source term for the loss of nitric acid containment accident is a function of the total amount available inside the storage tank (MAR), the amount released from the dumpster (DR), and the amount that drains into waste tank EP38. MAR and DR are terms typically used in determining the source term for radionuclide source terms, but they have applicability in the case of a chemical spill. The generalized equation for determining the quantity of material available for the source term is as follows:

$$\text{Dispersible Amount} = \text{MAR} * \text{DR} * (1 - \% \text{ drained}) \quad (\text{Eq. 3.4-23})$$

Once the available quantity of material is determined, the mass transfer rate is calculated to derive the amount of material evaporated in units of kilograms per second. S-CLC-H-00278 (Ref. 184) discusses the inputs to the mass transfer rate and the results of the calculation.

Material at Risk

The dumpster has a capacity of 1,000 gallons of nitric acid (Ref. 13). Accounting for a possible 10% head space in the dumpster that could be filled, the MAR equals 1,100 gallons of 50% nitric acid.

Damage Ratio

The DR is the fraction of the MAR that is impacted by the accident-generated conditions. For this scenario, it translates into the amount of material that would be released from the dumpster in the event of a breach or leak. This scenario assumes a total release of material. The DR is conservatively estimated to be 1.0.

Amount Drained to Waste Tank EP38

The dumpster is located on a bermed pad, which drains to waste tank EP38. The waste tank capacity is 6,400 gallons, and requirements call for draining the waste tank when 75% full, which allows for a minimum available capacity of

$$6,400 - (0.75 * 6,400) = 1,600 \text{ gallons.}$$

Since the nitric acid dumpster can never have more than 1,100 gallons of material, the waste tank can accommodate the entire contents of the nitric acid dumpster. However, to ensure a conservative estimate for the source term calculation, it is assumed that none of the nitric acid drains to the waste tank. Therefore, all nitric acid will be assumed to remain on the surface surrounding the dumpster.

Dispersible Amount

Based on the conservative estimates identified above, all 1,100 gallons are available for dispersion. This value is inputted into S-CLC-H-00278 (Ref. 184) to derive the evaporation rate.

Evaporation Rate

The mass transfer rate for unbermed, 1,100 gallons of 50% nitric acid is calculated in S-CLC-H-00278 (Ref. 184). The mass transfer rate is 0.0673 kg/sec.

This analysis assumes the nitric acid instantaneously forms a 1-cm deep pool, at a concentration of 70 wt%. The calculation takes no credit for the cooling of the material due to evaporation heat loss. In reality, the nitric acid source term generation rate over the initial 15 minutes of the event, where the addition of the neglected splashing source term has importance, would be substantially lower due to the order-of-magnitude partial pressure difference between 70 wt% and 50 wt% nitric acid. In addition, the temperature decreases due to evaporative cooling would further lower the nitric acid source term generation rate. The analysis is at least an order-of-magnitude higher than the realistic case and therefore the potential contribution of the source term due to splashing is more than accounted for in the conservatism of the calculation.

Total Source Term

The spill contribution to the source term is 0.0673 kg/sec of nitric acid (Ref. 184).

Consequence Analysis

The consequences for a 0.0673-kg/sec release of nitric acid (berm is not credited) use the methodology previously noted (Ref. 186) for the onsite receptors at 100 meters and 640 meters. The offsite receptor at 11.9 km is evaluated with the neutrally buoyant, Gaussian model stated in Equation 3.4-22, using the Briggs rural dispersion coefficients (table after Equation 3.4-22). In this case, for F stability, sigma-y at 11.9 km is 321.65 meters, and sigma-z at 11.9 km is 41.66 km. For the 1.5-m/sec wind speed, and a 0.0673-kg/sec source rate, the groundlevel concentration is 1.07 mg/m³, or 0.4 ppm (1 ppm = 2.62 mg/m³ for nitric acid).

As noted earlier, a neutrally buoyant model will overpredict concentrations (and is thus more conservative) at the MOI distances for the RBOF source coordinates (11 km to 12 km) relative to a dense gas model.

Concentration of Nitric Acid

Event	Release Rate (kg/sec)	Conc. @100m	Conc. @640m	Conc. @11.9km ²
Spill Without Credit for the Berm	0.0673	100 ppm	3.1 ppm	0.4 ppm

- a The RBOF is located in H Area. The minimum distance from the H-Area meteorological tower to the site boundary is 11.9 km (Ref. 130).

Comparison to Guidelines

Based on the event frequency for this event ($10^{-2}/\text{yr} > p > 10^{-4}/\text{yr}$), the appropriate concentration guidelines are <ERPG-2 and <ERPG-1 for an onsite and offsite receptor, respectively. The concentration guidelines for nitric acid are 15 ppm and 30 ppm for ERPG-2 and -3, respectively. The consequences for the receptor located 640 meters and at the site boundary are acceptable. (Refer to the hazardous chemical worker safety section of Section 3.3.2.3 for the onsite receptor consequence methodology and analysis.)

Summary of Safety-Class SSCs and TSR Controls

This accident scenario does not challenge the offsite EGs; therefore, no safety-class SSCs are required.

LOSS OF 30% SODIUM NITRITE CONTAINMENT

Loss of sodium nitrite containment will result in a chemical spill outside the facility. The hazards analysis categorized the risk of event 314a as having moderate consequences and an anticipated likelihood of occurrence, or a risk rank of 4.

Scenario Development

Sodium nitrite (NaNO_2), used as a waste tank corrosion inhibitor, is stored in a skid-mounted, 1,000-gallon dumpster (EP-503-100-1) on chemical pad No. 2, located to the west of Building 245-H. The contents of the dumpster are pumped to an adjacent holdup tank (EP373.65) with a maximum capacity of 1,600 gallons (Ref. 13). The holdup tank is also located on chemical pad No. 2 and is equipped with an agitator. The contents of the holdup tank are heated during winter months to approximately 10°F to 15°F above the crystallization point. The sodium nitrite dumpster is typically filled to 900 gallons and is administratively controlled to this volume to prevent exceeding the bermed area capacity. However, this analysis assumes that the contents of both tanks are filled to a total volume of 2,600 gallons. Chemical pad No. 2 is located approximately at ground level outside the RBOF and RRF and is surrounded by a dike.

The dumpster is shipped to another facility to be refilled with the appropriate chemicals. Preparation for shipping involves the disconnection of associated piping, flexible hoses, and instrumentation. Facility personnel follow the latest facility normal operating procedures while performing these operations.

Based on the hazards analysis, this accident analysis examines the frequency and consequences of a spill of 30% sodium nitrite from dumpster EP-503-100-1 and/or the adjacent holdup tank (EP373.65) due to causes such as material failure, mechanical damage, equipment malfunction, and human error. Sodium nitrite spills due to earthquake, severe weather, and surface vehicle impacts are covered under those respective accident analysis sections.

Chemical spills associated with filling (e.g., overfilling) of the dumpster outside of the RBOF and RRF are not considered here.

Possible Initiating Events

The following are possible initiating events for this scenario:

- Material failure of the dumpster or holdup tank or associated piping, valves, or flex hoses occurs due to fatigue or corrosion.
- Dumpster outlet valve (SN-3) fails to close due to mechanical failures.
- The dumpster is dropped during movement due to human error or failure of the lifting mechanism.

- Dumpster outlet valve (SN-3) fails to close due to human errors.

Accident Event Progression

If an initiating event occurs, the chemical would accumulate in the bermed area of the pad. The rate of leakage from the dumpster and/or holdup tank will depend on the point of breach and the severity of the opening. A worst-case breach would drain the entire contents of the dumpster and/or holdup tank within minutes.

For this scenario, no credit is given for the berm containing the chemical spill. Therefore, the sodium nitrite spreads over a large area, which results in a larger ARR than for a bermed release.

Preventive Features

A preventive feature for this scenario is that the dumpster outlet valve (SN-3) is normally shut.

Mitigative Features

The following are mitigative features for this scenario:

- The chemical storage dumpster and holdup tank are located in a bermed area that may contain liquids and may pump them to waste tank EP38. The waste tank volume is 6,400 gallons.
- The chemical pad is sloped to a center-mounted sump drain, which is equipped with a sump pump. This sump is connected through an isolation valve to an electric sump pump (LWS-P-001), which must be started manually. The pump discharge is routed to the hose box No. 2 drain, which gravity-drains to waste tank EP38. The drain is fabricated of Type 304 stainless steel. This drain ties into the header from chemical pad No. 1 and the hose rack leading to the waste tank.
- Sodium nitrite has an attraction to water vapor. Therefore, the sodium nitrite will never completely dry out but rather will form a firm, moist pack of NaNO_2 .

Frequency Binning

A fault tree was developed to quantify the frequency of spilling sodium nitrite due to the initiating events listed above. The fault tree analysis is documented in S-CLC-H-00281 (Ref. 179). The frequency of the event, as estimated by the fault tree, is $5.69\text{E-}03/\text{yr}$, which is in the unlikely ($10^{-2}/\text{yr} > p > 10^{-4}/\text{yr}$) frequency bin.

The fault tree was developed and quantified using the CAFTA computer code (Ref. 180). Input data for each basic event, consisting of component failure rates and human error probabilities, are

obtained from NUREG/CR-4639, NUREG-0492, DP-1633, and NUREG/CR-1278-F (Ref. 107, 108, 181, 182). The logic of the fault trees is arranged into four major categories: material failure, equipment malfunction, mechanical damage, and human error.

The following assumptions were used as input and are specific to the fault tree evaluation performed for a nitric acid spill:

- The facility is operated 24 hours a day, 365 days a year. Dumpster EP-503-100-1 is always be in operation, except when removed from the system for refilling.
- Dumpster EP-503-100-1 and holdup tank EP373.65 combined always contain at least 167 gallons of sodium nitrite. A large chemical spill, from dumpster EP-503-100-1 and the adjacent holdup tank EP373.65 combined, is defined in the RBOF hazards analysis as a spill greater than 167 gallons of 30% sodium nitrite. A spill greater than 167 gallons will result in exceeding ERPG limits for a hazardous chemical release to the environment. A large chemical spill from the dumpster or holdup tank results in the spill magnitude of concern. **(Credit assumed for Hazardous Material Inventory Program)**
- Dumpster EP-503-100-1 and holdup tank EP373.65 are stored on chemical pad No. 2. The top surface of this pad is installed essentially at ground level. A dike is installed around the perimeter of the chemical pad to contain any leaks from the dumpster and associated piping systems. Success or failure of this dike to collect or contain the spill is not considered when determining the frequency of a sodium nitrite spill.
- Chemical pad No. 2 contains a sump and associated pump to direct any chemical spill to waste tank EP38. No credit is taken for this system when determining the frequency of a sodium nitrite spill.
- Dumpster EP-503-100-1 is typically filled approximately 15 to 18 times a year. The maximum number of refills anticipated for the dumpster is 24 times a year.
- No credit is taken for any inspection or testing performed on the truck-mounted uni-arm swing-out lever used for lifting the dumpsters from the chemical pads.
- A dumpster (or attached piping or valve) is breached when the dumpster is dropped during a loading/unloading operation. No credit is taken for any vessel certification to DOT standards or drop-test qualifications.
- No credit is taken for independent checking performed on proceduralized activities involving connection and disconnection of the dumpsters on chemical pad Nos. 1 and 2.
- The following piping/hose lengths are assumed for this analysis:
 - A cumulative total of 20 feet of piping is connected between the sodium nitrite dumpster and the holdup tank. The underground piping between the holdup tank and the valve room is not considered in this evaluation; however, there is

approximately 1 foot of piping between the holdup tank and the tank outlet valve which is considered.

- A total of 5 feet of flexible hose is connected between each dumpster and the respective metering pump inlet valve.

These sections of pipe/hose are conservatively assumed to be constantly exposed to fluid and, therefore, always vulnerable to a large spill. Other "downstream" piping or valves are not considered since they are vulnerable only when transfers are in progress, which is only a small fraction of a year. Therefore, failures of these components are considered insignificant contributors to the overall spill frequency.

- Failure due to structural fatigue is not considered because the dumpsters and associated piping and valves are not subjected to cyclic structural loading in excess of the fatigue limit (i.e., 2 to 10 million cycles) for most steels (Ref. 183). Although corrosion due to caustic and acidic chemicals can substantially reduce fatigue strength, the low frequency of valve manipulations involving the connection and disconnection of hoses and vessel movements associated with refilling activities (i.e., once per week at worst case) does not warrant analysis for corrosion fatigue. The dumpsters are vented to the atmosphere (i.e., nonpressurized vessels).
- Failure due to thermal stress is not considered because the dumpsters and associated piping and valves are not normally subjected to cyclic high temperature transients. The equipment is exposed to ambient conditions.

The dominant scenario for a potential sodium nitrite spill is due to a dropped dumpster resulting in mechanical damage breaching the container.

Source Term Analysis

The source term for the loss of sodium nitrite containment accident is a function of the total amount available (MAR), the amount released from the tank and dumpster (DR), the amount that drains into waste tank EP38, and the rate of airborne release. MAR and DR are terms typically used in determining the source term for radionuclide source terms, but they have applicability in the case of a chemical spill. The generalized equation for determining the source term that is evaporated and available for dispersion is Equation 3.4-20.

Because NaNO_2 is a salt, as the water content of sodium nitrite evaporates, most of the chemical will accumulate on the pad. Only a small fraction, based on the ARR, will be dispersed as part of the evaporating water.

Material at Risk

The dumpster and tank have a combined capacity of 2,600 gallons of 30% sodium nitrite. The sodium nitrite dumpster is typically filled to 900 gallons, and the total allowable of 900 gallons is

administratively controlled. However, this analysis assumes that both tanks are filled to a total volume of 2,600 gallons.

Damage Ratio

The DR is the fraction of the MAR that is impacted by the accident-generated conditions. For this scenario, the DR translates into the amount of material that would be released from the tank in the event of a breach or leak. This scenario assumes a total release of material. The DR is conservatively estimated to be 1.0.

Amount Drained to Waste Tank EP38

The tank is located on a bermed pad, which drains to waste tank EP38. The waste tank capacity is 6,400 gallons, and requirements call for draining the waste tank when 75% full, which allows for a minimum available capacity of

$$6,400 - (0.75 * 6,400) = 1,600 \text{ gallons.}$$

Since the sodium nitrite tanks might contain 2,600 gallons of material, the waste tank can accommodate 62% of the sodium nitrite tank volume. However, to ensure a conservative estimate for the source term calculation, it is assumed that no material drains to the waste tank. Therefore, all sodium nitrite will remain on the surface surrounding the dumpster.

Airborne Release Rate

The sodium nitrite will accumulate around chemical pad No. 2 (no credit is taken for the berm) and will be entrained into the air. Since sodium nitrite is a salt solution, the water content of the chemical will evaporate while the salt component will accumulate and concentrate in the bermed area. A small fraction of the sodium nitrite particulate will disperse with the evaporating water. This fraction, the ARR, is provided in DOE-STD-3010-94 (Ref. 75). Its value is

$$ARR = 4E-06/\text{hr.}$$

Total Source Term

Based on the discussion above, the rate of sodium nitrite evaporation can be determined as follows:

$$\begin{aligned} \text{Source Term} &= 2,600 \text{ gallons} * \text{DR} * \text{ARR} * \text{duration} && (\text{Eq. 3.4-24}) \\ &= 2,600 * 1.0 * 4E-06/\text{hr} \\ &= 1.04E-02 \text{ gal/hr of 30\% sodium nitrite} \\ &= 1.04E-02 \text{ gal/hr} * (1 \text{ hr}/3600 \text{ sec}) * (3.033 \text{ lb} \cdot \text{NaNO}_2/\text{gal}) * (1 \text{ kg}/2.205 \text{ lb}) \\ &= 4E-06 \text{ kg/sec of NaNO}_2 \text{ disbursed to the air.} \end{aligned}$$

Consequence Analysis

The same methodology used to determine the concentration of sodium hydroxide is applied here. The airborne source term, Q , is $4\text{E-}06$ kg/sec. The wind speed, u , is assumed to be 4.5 m/sec for the onsite consequences. The lateral and vertical dispersion parameters, σ_y and σ_z , use a D stability class and a downwind distance of 100 meters. The parameters are therefore 8.0 meters and 5.6 meters, respectively.

Incorporating the appropriate values into the Gaussian plume equation, the concentration is $6\text{E-}03$ mg/m³.

Comparison to Guidelines

The lowest concentration guideline (PEL-TWA) for either onsite or offsite and independent of event frequency for sodium nitrite is 10 mg/m³ (Ref. 186). Therefore, the consequences of the release (2,600 gallons) are insignificant, and there is no need for further analysis at greater distances.

Summary of Safety-Class SSCs and TSR Controls

This accident scenario does not challenge the offsite EGs; therefore, no safety-class SSCs are required.

3.4.2.2 Surface Vehicle Impact

Surface vehicle impact with a chemical dumpster was identified as an event to be analyzed in the hazards analysis. The events in the hazards analysis that identified this initiator are Events 314a, 316, and 318, having moderate consequence and an anticipated likelihood of occurrence, or a risk rank of 4. Events 345c and 345d also address this potential scenario and are categorized with a risk rank of 5. The consequences of these three events are the same as the consequences for the events analyzed in Section 3.4.2.1. The difference is that this scenario only evaluates one initiating event, surface vehicle impact, whereas Section 3.4.2.1 evaluated "internal" initiators. Therefore, while the frequency of occurrence is expected to be different, the source term will remain the same as is calculated in Section 3.4.2.1.

SCENARIO DEVELOPMENT

Three chemical dumpsters are located outside of the RBOF building on chemical pads. These are a 1,000-gallon 50% nitric acid dumpster and a 1,000-gallon 50% sodium hydroxide dumpster, both located on chemical pad No. 1, and a 1,000-gallon 30% sodium nitrite dumpster and associated 1,600-gallon holdup tank located on chemical pad No. 2. The sodium nitrite volume is administratively controlled to 900 gallons, but this analysis assumes the physical

allowable volume of 2,600 gallons. The dumpsters are replenished with chemicals by disconnecting the piping and transferring the dumpster to F Area for filling. The dumpsters are then returned to the RBOF and re-connected.

In this scenario, a vehicle operator loses control and impacts one of the chemical dumpsters and/or tanks. An impact is likely to dislodge the dumpster/tank and cause the piping to crack and spill the chemical contents of the dumpster/tank.

POSSIBLE INITIATING EVENTS

For this scenario, there is only one initiating event, a surface vehicle impact.

ACCIDENT EVENT PROGRESSION

In the event that a vehicle were to impact a chemical dumpster/tank, it is possible that the dumpster/tank would be breached, or the associated piping could separate from the vessel causing the contents of the vessel to spill into the bermed area. It is unlikely that impacts would occur at "high" speeds since there is a 10-mph speed limit and the area is congested. Therefore, an associated vehicle fire is considered remote and the only consequence would be from the spilled chemical as it evaporated and dispersed into the air.

PREVENTIVE FEATURES

The following are preventive features for this scenario:

- The H-Area security fence, a steam line, and a storage building are barriers between the chemical pads and the main roadway.
- There are berms surrounding the chemical pads that could act to prevent vehicle access to the dumpster/tank.

MITIGATIVE FEATURES

The following are mitigative features for this scenario:

- The chemical pads are located in bermed areas that either drain directly to or may be pumped to waste tank EP38. Structural analysis of the berm has determined that the berm can withstand the lateral loads associated with the impact of a truck wheel. The forces placed on the berm are well within the berm's capacity without cracking (Ref. 73, 187).
- The speed limit inside the security fence is 10 mph.

- If the dikes surrounding the chemical pads do not prevent a vehicle from accessing the pad, it will at least impede the travel of the vehicle.

FREQUENCY BINNING

The frequency determination for surface vehicle impact at the RBOF and RRF is based upon topography, area layout, administrative controls, and frequency of vehicle traffic. The frequency is estimated as $2.90\text{E-}04/\text{yr}$, which is in the unlikely ($10^{-2}/\text{yr} > p > 10^{-4}/\text{yr}$) frequency bin. The analysis is documented in S-CLC-H-00281 (Ref. 179) and is summarized as follows.

The frequency of surface vehicle impacts into the chemical pads has two components: (1) the frequency due to impacts from vehicles originating outside of the security fence, and (2) the frequency due to impacts from vehicles originating inside of the security fence.

The frequency contribution due to impacts from vehicles originating outside of the facility is considered insignificant due to the following reasons:

- The security fence surrounding the RBOF and RRF acts as a barrier for the chemical pads.
- The road is at a lower elevation than the chemical pads, which would result in a reduction of vehicular speed.
- The distance from the road to the chemical pads represents a stretch of ground for slowing of any vehicle that might accidentally penetrate the security fence.
- There are limited direct lines of approach to the chemical pads (barriers such as a steam pipe and a storage building are located between the dumpsters and the main roadway).

Therefore, the frequency for surface vehicle impact into the chemical pads is based on vehicles within the H-Area security fence in the vicinity of the RBOF and RRF. At a minimum, this traffic may consist of government and security, construction, service and maintenance, operations, commercial vendor, and routine delivery vehicles.

It is assumed that the maximum number of vehicles entering the RBOF/RRF area is 50 per day (or 18,250 per year). Further, it is assumed that at least half of this traffic enters the east side of the RBOF, exits by the same service road, and has no need to enter the area of the chemical pads located on the west side of the RBOF and RRF. Traffic entering the west side near the chemical pads consists primarily of a limited number of government vehicles, hourly security checks by Wackenhut, service and maintenance vehicles, deionizer/dumpster deliveries, and a limited number of contract personnel. Thus, the maximum frequency of vehicle traffic entering the area of the chemical pads is taken to be 9,125 vehicles per year.

The frequency of vehicle accidents is taken from a similar study for the Solid Waste Management Facility (SWMF) (Ref. 121). These frequencies (based on accident severity) are as follows:

Accident Severity Class	Accident Frequency/Vehicle Mile
Minor	6.3E-06
Moderate	1.2E-07
Severe	1.9E-08
Extra Severe	9.0E-09
Extreme	3.2E-09

Damage to the chemical pads that results in a large chemical spill is assumed to involve accident severities of at least a moderate class. Therefore, the frequency of a moderate or greater accident is obtained by summing the frequencies for a moderate, severe, extra severe, and extreme accident severity. The resultant frequency is 1.51E-07 per vehicle-mile.

The distance around the RBOF and RRF is taken to be 0.21 miles (estimated from site maps). Multiplying this value by the estimated number of vehicles traveling per year in the vicinity of the chemical pads (9,125) results in 1,917 vehicle-miles/yr.

Therefore, the frequency of surface vehicle impacts into the chemical pads that result in a large chemical spill is calculated as 1.51E-07/vehicle-mile times 1,917 vehicle-miles/yr, or 2.90E-04 occurrences/yr.

SOURCE TERM ANALYSIS

The source terms for the three possible chemical spills, 50% nitric acid, 50% sodium hydroxide, and 30% sodium nitrite, are the same as in Section 3.4.2.1. Although the release due to a vehicle impact may be slightly more energetic, the leak rate and the total size of the spill will probably be smaller. Therefore, the scenario assumptions used to calculate the source terms in Section 3.4.2.1 are also assumed herein.

Section 3.4.2.1 indicates that sodium hydroxide will be disbursed at a rate of 3.5E-06 kg/sec and that 6.73E-02 kg of nitric acid will evaporate per second in an unbermed spill. In addition, Section 3.4.2.1 indicates that sodium nitrite will evaporate at a rate of 4E-06 kg/sec.

CONSEQUENCE ANALYSIS

The consequences are the same as those presented in Section 3.4.2.1. For the sodium nitrite and sodium hydroxide releases, the consequences of the release are insignificant and there is no need for further analysis at greater distances. For the release of nitric acid, the consequences for the

receptor located 640 meters and at the site boundary are acceptable. (Refer to the hazardous chemical worker safety section of Section 3.3.2.3 for onsite receptor consequence methodology and analysis.)

COMPARISON TO GUIDELINES

Based on the frequency for this event, the comparison to EGs is based on the analysis in Section 3.4.2.1.

SUMMARY OF SAFETY-CLASS SSCs AND TSR CONTROLS

This accident scenario does not challenge the offsite EGs; therefore, no safety-class SSCs are required.

3.4.2.3 Process-Induced Criticality

The hazards analysis identified the following 10 events as process-induced criticalities:

- Event 34, floating fuel container
- Event 35, fuel storage infraction
- Event 37b, deformation of the fuel storage rack
- Event 38, critical configuration in a fuel bundle
- Event 39, configuration of fissile debris on basin floor
- Event 40, fuel spills from fuel basket during fuel movement
- Event 40a, shipping cask contents are spilled or cask lid is dropped
- Event 41, more than one basket out of cask at same time resulting in criticality
- Event 42a, 70-ton cask drains
- Event 44, disassembled pieces are arranged in an unsafe geometry on basin floor

This analysis combines these different potential events into a single fault tree and uses a single, bounding fission yield to define the upper bounds of the consequence.

SCENARIO DEVELOPMENT

As part of its mission, the RBOF receives and stores a wide variety of nuclear fuel assemblies. Handling includes removal of assemblies from shipping containers, placement of fuel assemblies into shipping containers, placement in the storage pool, and movement around the storage pool. The fuel assemblies are moved by means of a traveling bridge/monorail system and are stored in

lockable storage racks separated at critically safe distances. This accident scenario involves a criticality event due to fuel handling or storage infractions.

POSSIBLE INITIATING EVENTS

Due to the nature of operations performed in the RBOF, there are many events that have the potential to lead to a criticality. These mechanisms include the following:

- Human errors during the placement of fuels into storage
- Human errors during the loading of fuel bundles
- Human errors during relocation of fuel in storage
- Human errors during movement of fuel between storage and the working basins
- Human errors in moving the traveling bridge hoist, which catches on a rack and deforms the rack
- Human errors during loading of the transfer bucket during cask unloading operations
- Dropping fuel bundles, assemblies, or transfer baskets
- Movement of fuel containers to within 12 inches of one another
- Overloading a shipping cask
- Human errors during assembly cropping operations
- Floating fuel containers
- Breached fuel containers
- Human errors during movement of the shipping cask or cask lids
- Equipment failures

ACCIDENT EVENT PROGRESSION

Assuming that a critical configuration is achieved, the storage pool water would act to reduce the event's source term. Because this particular scenario evaluates a range of initiating events, the specific steps in the sequence between initiating event and the criticality are not detailed. The criticality event is assumed to result in a $5E+17$ fission yield, as discussed in Section 3.4.1.5.

PREVENTIVE FEATURES

Measures to prevent a criticality at the RBOF include the following:

- Storage racks are configured and spaced to prevent criticality events. (Credited)
- NSDSs specify safe spacing distances between fuel bundles. (Credited)

- Workers are trained in the safe operation of the traveling bridge and safe handling of fuel assemblies. (Credited)
- Administrative controls are placed in operating procedures.

Chapter 6 provides further discussion regarding these events and available preventive features.

MITIGATIVE FEATURES

The following are mitigative features for this scenario:

- Fuels are stored in water-filled pools under a minimum of 4-1/2 feet of water for scrubbing of fission products and attenuation of direct dose from criticality. (Credited)
- The building exhaust filters further reduce the particulate source term.
- The NIMs alert operators that a criticality event with a fission product release is in progress. (Credited)
- Emergency response plans and onsite medical treatment personnel facilitate removal and treatment of injured workers.
- Site audible warning systems alert workers of the need to evacuate the area.
- Workers are trained to respond safely in the event of a criticality. (Credited)

FREQUENCY BINNING

A fault tree was developed to quantify the frequency of a criticality in the RBOF facility. The fault tree analysis is documented in S-CLC-H-00340 (Ref. 188). The fault tree quantifies the overall frequency of a criticality including an NPH-induced criticality. The frequency of criticality due to initiators other than natural phenomena is estimated to be $6.64\text{E-}04/\text{yr}$, which is in the unlikely ($10^{-2} > p > 10^{-4}$) frequency bin.

The fault tree development for events other than natural phenomena is two-fold: (1) criticality due to process-related initiators, and (2) criticality due to miscellaneous initiators. These initiators are discussed separately below.

Process-Related Initiators

Process-related criticality is defined as criticality events caused by the planned movement/manipulation of fuel in which infractions are made to the point that a critical mass of fuel is brought into a geometry favorable for criticality. The approach made in the fault tree to quantify the criticality frequency due to these types of errors is to model the types of operations performed in the RBOF, in terms of number of operations per year, and couple this with errors

that could be made that could potentially bring a critical mass of fuel together into a geometry favorable for criticality. The types of operations considered in the fault tree are as follows:

- Storage basin operations
 - Storing fuel from cask unloading
 - Performing traveling bridge operations
 - Relocating fuel within the storage basin
 - Transferring fuel from the storage basin to a working basin
- Cask unloading operations
 - Preparing the cask for fuel removal
 - Retrieving fuel from the cask
 - Loading transfer buckets
- Cask loading operations
 - Preparing the cask for fuel addition
 - Retrieving fuel from storage for loading
 - Loading fuel
- Working basin operations
 - Cropping assemblies
 - Placing buckets in isolation
 - Loading bundles

The number of operations performed on a yearly basis are intended to be bounding and are as follows: (Ref. 59)

- The number of casks unloaded at the RBOF on a yearly basis is 112.
- The number of LWT type casks unloaded at the RBOF on a yearly basis is 40.
- The number of LWT type casks unloaded at the RBOF containing cask baskets loaded with a number of assemblies exceeding K_{safe} on a yearly basis is 10.
- The number of bundles loaded in a SRS 70 ton cask is nominally 15.
- The number of transfer buckets used per cask to unload and transfer fuel is no greater than 33.
- The number of casks loaded at the RBOF on a yearly basis is 60.
- The number of assemblies added to the RBOF inventory on a yearly basis is 3,600.
- The number of fuel relocation jobs performed within the storage basin at the RBOF on a yearly basis is 36.

- The number of fuel bundles relocated on a yearly basis is 500.
- The number of partially filled bundles retrieved from the storage basin to "fill" at the RBOF on a yearly basis is 90.
- The number of assemblies cropped on a yearly basis is 2,000.
- The number of HFIR cores unloaded at the RBOF on a yearly basis is 30.

The potential errors that could be made during fuel handling operations are based upon procedure review and existing NSDSs. All fuels received in the RBOF are required to have an NSDS, which specifies such items as the safe number of fuel units permissible in the worst configuration, safe spacing along rows and between adjacent rows, and other fuels that it may interact within adjacent rows.

The types of errors modeled in the fault tree are typically those in which too much fuel is brought together in violation of the NSDS (e.g., overloaded fuel bundle or transfer bucket), inadequate spacing is maintained (e.g., failure to place blank bundle between fuel bundles as required), fuel is stored in the wrong place (i.e., violation of interaction controls), or fuel is dropped. In general, a single error is modeled (e.g., a blank bundle is not placed between fuels in a row as required); however, there are typically multiple mishaps required to occur to result in the criticality.

There are exceptions where multiple errors are modeled (e.g., two or more assemblies added to a bundle in the working basin). The reason multiple errors are modeled in some places and single errors in others is based on the analysis that was performed to establish the limits. The safe number of units in worst-configuration calculations was based on conservative analyses. The single errors are modeled as such when there is no basis (based on prior calculations) for assuming that the error by itself is necessarily safe.

The result of this effort is a logic model that, using the scenarios identified in the hazards analysis as a starting point, identifies situations that could potentially cause a criticality due to a critical mass of fuel forming a critical geometry.

The next phase of this analysis involved adding conditional probabilities to the fault tree logic to account for whether the scenario actually results in a criticality. The scenarios of concern were specified in ECS-RAG-950071 (Ref. 146). The specified charter was to examine the reactivity characteristics of certain types of fuel within the RBOF and examine the specific scenarios identified in ECS-RAG-950071 to determine whether or not they are critical.

A plan for completing the criticality accident analysis (Ref. 189) was developed in which it was determined that only those fuels believed to be the most reactive would be examined for reactivity characteristics. These fuels are as follows:

- MURR fuel
- MIT fuel
- HWCTR fuel

- SFO
- HFBR
- ERR
- SRE
- Dresden
- HFIR

Due to the high reactivity of HFIR cores and to the uniqueness of the operational procedures developed for use in handling those elements, specific scenarios were included in the fault tree related to receipt, storage, and handling of HFIR fuel.

The approach is to focus only on those fuels that are assumed to be most likely to cause criticality concerns and to assume for the fault tree that all fuel handling operations involve those fuels. Future plans are to examine remaining fuels as needed to support the USQE process. It is assumed that basing the fault tree analysis solely on those fuels identified above provides a bounding frequency with respect to causes other than natural phenomena. In other words, it is believed that if future operations in the RBOF were modeled exactly, a lesser frequency of criticality would be obtained, because a significant fraction of the operations performed would involve fuel less reactive than those considered here. The methodology for determining the reactivity characteristics for the examined fuels is given in Section 3.4.1.4.

The logic developed in the fault tree accounts for those scenarios that have the potential for a criticality due to infractions during movement/manipulation of fuel, coupled with conditional probabilities that the infractions actually cause criticality based on the fuel reactivity calculations. In general, the conditional probabilities are set at 1.0 if the reactivity calculations indicate a criticality could result, or 1E-06 if the reactivity calculations dismiss the possibility of a criticality. Specifically, the scenarios that were either deemed to be a criticality scenario, or assumed to be a criticality scenario, and their respective frequencies are as follows (Ref. 188):

- The hoist hook/tool on the traveling bridge crane snags and deforms a rack - 1.66E-05/yr
- Infractions during retrieval of fuel from storage basin - 5.64E-08/yr
- Incorrectly loaded bundle stored incorrectly - 2.29E-05/yr
- Incorrectly loaded bundle stored correctly - 8.28E-08/yr
- Correctly loaded bundle stored incorrectly - 1.45E-04/yr
- Cask unloading infractions - 1.63E-04/yr
- Cask loading infractions - 1.86E-05/yr
- Fuel relocation infractions - 2.11E-05/yr
- Working basin infractions - 6.89E-05/yr

- Scenarios involving HFIR cores - $1.60\text{E-}04/\text{yr}$

Miscellaneous Initiators

The logic provided in the fault tree for these initiators quantifies the frequency of criticality due to causes other than natural phenomena, and other than planned movement/manipulation of fuels. Specifically, the hazards analysis identified two scenarios that are covered here: (1) floating fuel container results in criticality, and (2) configuration of fissile debris on basin floor causes criticality.

The logic was developed for the scenario, and as discussed under the subsection on process-related initiators, conditional probabilities were provided based on reactivity characteristic calculations for fuels believed to be the most reactive. The conditional probabilities are set at 1.0 if the reactivity calculations indicate that a criticality could result, or $1\text{E-}06$ if the reactivity calculations dismiss the possibility of a criticality. Specifically, the floating fuel container was deemed to not cause a criticality, whereas a critical configuration of fissile debris can be caused by breached fuel containers at a frequency of $7.60\text{E-}06/\text{yr}$.

The following assumptions were made during the fault tree analysis:

- The analysis of criticality frequency due to causes other than natural phenomena is based largely on the number of fuel movement operations on a yearly basis, coupled with nuclear safety control infractions. To determine what combination of infractions could actually result in a criticality, a parallel effort was made to determine the reactivity characteristics of what is believed to be the most reactive fuels in the RBOF (Ref. 189). The fault tree analysis is based on the results of these reactivity studies (Ref. 151 through 165), and it is assumed that all fuels handled in the RBOF are similar to these types of fuel. Therefore, based on the reactivity characteristics of these fuels, scenarios are either deemed to cause a criticality or deemed to not cause a criticality. It is assumed that basing this analysis on only these fuels provides a bounding frequency with respect to causes other than natural phenomena.
- Two assemblies are placed into a bundle or bucket during applicable fuel handling operations for the purposes of determining the number of these fuel handling operations performed on a yearly basis. Two is a lower bound and therefore provides a conservative estimate.
- A conservative estimate for the number of transfer buckets that fuel is loaded into during cask unloading is not greater than 15.
- Proper inserts may be required in the transfer buckets according to procedure.
- NUREG-0492 (Ref. 108) provides a generic human error database that is applied almost exclusively in the fault tree. The data give low, nominal, and high human error probabilities for a variety of human errors. These data are assumed applicable to this analysis, and a low, nominal, or high value for the various errors is assigned

based upon analyst judgment concerning conditions leading up to and surrounding the event. Other sources of data for this analysis are given in References 151 through 165. There are assumptions/judgments related to selection of the probability for many of the basic events modeled in the fault tree, and these assumptions/judgments are clearly identified in Appendix A of S-CLC-H-00340 (Ref. 188).

SOURCE TERM ANALYSIS

A criticality will produce fission products and energy. A portion of this energy will be deposited in the fuel in the form of heat. It is this heat that can lead to the melting of the fuel and clad. Supporting documentation (Ref. 90) suggests that, if the criticality fission yield is below $1\text{E}+18$, the probability of the fuel melting is low (1 in 10 chances, or 0.1). For this analysis, the fuel is assumed to melt at the assumed frequency of the criticality with no reduction due to this conditional probability. For the process-induced criticality events, it is conservatively assumed that all of the released fission energy would go towards adiabatically heating of the fuel. Heat is the mechanism both for creating a leak path for the source term (clad melting) and for volatilizing a portion of the fuel (fuel melting). The escaping volatile and nonvolatile isotopes represent the source term for the criticality.

The MAR from a $1.0\text{E}+20$ fission yield was developed in N-CLC-H-00138 (Ref. 190) as shown in Table 3.4-5. That MAR is linearly scaled down to a $5.0\text{E}+17$ fission yield and is presented in Table 3.4-3. DRs, ARFs, and LPFs were then applied to the MAR to determine the source term. The source term for this event is fully detailed in S-CLC-H-00399 (Ref. 191). The source term is represented by the following equation:

$$ST = \sum_i (MAR_i * DR_i * ARF_i * LPF_i) \quad (\text{Eq. 3.4-25})$$

where i represents the particular isotope.

Material at Risk

The MAR for the criticality contains both radionuclides originally in the fuel and those created during the criticality. In an earlier work, the fission products for $1\text{E}+20$ fissions were calculated using ORIGEN (Ref. 190). This unscaled fission product MAR is given in Table 3.4-5. These values must be scaled by a factor of 0.005 for the $5\text{E}+17$ fissions expected (Table 3.4-3).

The MAR for the radionuclides originally in the fuel is bounded by the number of RFAs that could possibly melt during the event. The RFA is a hypothetical bundle (assembly) that was developed to represent a bundle that would produce a bounding consequence for RBOF accident analysis (Ref. 15, 21). To determine the number of involved assemblies, a conservative analysis was performed which assumed that all of the released fission energy would go towards adiabatically heating the fuel (Ref. 191). It is this heat that leads to the melting of the fuel

cladding and fuel meat. Heat is the mechanism both for creating a leak path for the source term (clad melting) and for volatilizing a portion of the fuel (fuel melting).

Using the fuel and cladding characteristics of the MURR and MIT fuels and the conservative inventory of the RFA, the calculation estimates that 3.71 RFAs will melt during a $5E+17$ fission yield event. Therefore, the amount of isotopes available for release from the melted fuel is determined by the inventory of an RFA multiplied by 3.71 (Table 3.4-4).

ADJUSTMENT FACTORS TO THE SOURCE TERM

In a criticality event, two factors will mitigate the release. The first is the ARF, which is the amount of material that will escape from the damaged fuel into the water. The second factor is the leak path factor, which is the percentage of isotopes that will escape from the water and become airborne. No credit is taken for the building exhaust filters and the exhaust ventilation system.

In DOE-HDBK-3010-94 (Ref. 75), Mishima suggests a value of 0.1 for the DR for radionuclides generated by a criticality involving metal pieces. For the purposes of this calculation, the DR will be assumed to be 1. The ARFs used come directly from Table 3.4-3 (Table 6-10 of DOE-HDBK-3010-94).

The LPFs specified in DOE-HDBK-3010-94 define the amount and type of isotopes expected to escape the basin water (once released from the melted fuel) and become airborne from the basin water surface. The values associated with the noble gases and halogens (Table 3.4-3) are not dependent upon water depth; consequently, the LPF is specified as 1.0, or 100% of the released isotopes from the melted fuel become airborne. For the remaining groups of chemical classes, the amount of isotopes released is strongly dependent upon several factors, including the amount of water covering the fuel which is assumed to melt, the amount of basin water subcooling, particulate size, and the carrier gas-steam mass fraction (Ref. 99).

According to Reference 99, DFs (or $1/LPF$) on the order of 25 to 30 are achieved simply due to the pool being subcooled. Combined with a pool depth of 6 meters (19 feet), this could lead to DFs for particulates on the order of 20,000. For the RBOF analysis, a DF of 2,000 ($1/5E-04$) is assumed for conservatism due to the potential variation in the depth of basin water covering the postulated criticality location and for consistency with the previous minimum basin levels associated with -19 inches (below the floor level) defined in past safety analyses (Ref. 46). This reduction in the assumed DF also accounts for basin water subcooling. Controls currently exist for maintaining the basin water temperatures at or below 40°C. Subcooling is defined in Reference 99 as 27°C, which for the RBOF may be exceeded at times. Therefore, the reduction in DF from a suggested value of 20,000 to an assumed value of 2,000 would account for any conditions where subcooling cannot be ensured.

The ARF and LPF values are taken from DOE-HDBK-3010-94 and are presented in Table 3.4-3. The overall reduction factors (which include the DR, ARF, and LPF) appear in Table 3.4-2.

SOURCE TERM

The source term associated with the inventory in the basin (fuel melting) is given in Table 3.4-4. The total source term is then arrived at by adding to the product of the fission products (Table 3.4-5), fission scale factor (0.005), and the overall release fractions (Table 3.4-2).

Process Error-Induced Criticality Consequence Analysis

Radiological consequences resulting from a process error-induced criticality are calculated for the MOI and the individual at the OEP using the methods described in Section 3.4.1 and are summarized in Table 3.4-14. Source terms are derived as described above. Results are substantiated in S-CLC-H-00405 (Ref. 192).

As an alternative to the bounding consequence analysis performed above, the following analysis is provided to illustrate the methodology (Section 3.4.1.6) for propagating uncertainties in a dose calculation. The information presented in this section is contained in S-CLC-H-00602 (Ref. 229). The resulting doses from this alternative methodology and analysis will be used for comparison to the established EGs in lieu of the bounding doses previously established.

Dose Calculation

The dose is estimated from the following equation and summed over the significant isotopes:

$$\text{Dose} = \text{FIS} * \text{EGY} * (1/\text{EMELT}) * \text{INV} * \text{ARF} * \text{LPF} * (\chi/Q) * \text{B} * \text{DCF} \quad (\text{Eq. 3.4-26})$$

where:

Dose	=	inhalation dose (rem effective dose equivalent)
FIS	=	number of fissions
EGY	=	3.28E-17 MJ/fission
EMELT	=	energy to melt a fuel assembly (MJ)
INV	=	inventory of the isotope in an assembly (Ci)
ARF	=	fraction of the isotope released from the fuel on melting
LPF	=	fraction of isotope not removed by the water cover
χ/Q	=	dispersion factor to site boundary (sec/m ³)
B	=	standard man breathing rate of 1.25 m ³ /hr (=3.47E-04 sec/m ³)
DCF	=	dose conversion factor (rem per Ci of the isotope inhaled)

Parameter Values and Uncertainty Range

Several of these parameters have substantial uncertainty as follows.

Fission Burst (FIS)

The number of fissions in criticality events with water moderated solid fuel varies substantially. A log normal distribution fit of the data for internal events appropriate for process upset conditions data (see EPD-CTG-950067 and S-CLC-H-00343 [Ref. 77, 175]) indicates a median of $FIS=1.1E+17$ fissions with a 95% uncertainty factor of 4.5 for criticalities that could be characteristic of process upset conditions. (In energy units, this corresponds to a 50th percentile or median fission burst energy of 3.8 MJ and a 95% upper bound of 17.3 MJ).

Energy to Melt Assembly (EMELT)

The energy-to-melt is amount of sensible heat required to bring the fuel element/assembly to its melting point, and is a function of fuel types. The majority of fuel in RBOF is metallic U-Al fuel.

Metallic (U-Al) type fuel must be heated up into the 600°C range before melting and release of radioisotopes, and even to higher temperatures for oxide fuels. The fission energy required to melt a fuel assembly also depends on how fast the energy is generated in comparison with heat transfer from the fuel matrix to the clad and coolant. A best estimate is the energy to heat both the fuel and clad 565°C. From the table below (which shows the energy to heat the fuel in one storage slot for three fuel types) this would average about 10 MJ. In a slow transient this would be even higher due to heat loss to the surrounding water. Furthermore, since the energy generation in the fuel participating in the criticality will not be uniform, some of the fuel will be heated more than 565°C and some less than 565°C. This would decrease the number of assemblies that can be heated 565°C with a given heat input.

The maximum number of assemblies that could melt from a given energy deposition is that with the lowest energy to melt, which for the MIT fuel (conservatively low heat capacity fuel) is calculated to be 4.3 MJ. (See table of masses below, which is based on information in References 151, 152, and 200 and the assumption that the storage location would accommodate 6 MURR or MIT assemblies.). A lower energy of 0.85 MJ per assembly could theoretically result if it were assumed all the energy remains in the fuel meat. However, because of the very thin (0.01 to 0.02 inch) fuel layer thicknesses in this fuel having a fuel to clad thermal time constant on the order of 1 msec this is not possible even in a very rapid transient (3 to 5 msec periods) similar to those achieved in the SPERT super prompt critical burst tests.

The lower bound is assumed to be heating the fuel meat and inner half of the clad thickness which is calculated to require a total of 2.6 MJ per assembly for the MIT fuel (most limiting) (Note that for the Mark 42s, which have a much longer fuel to clad time constant, the energy to

heat only the fuel meat is significantly greater than this). Thus the median value for EMELT=10 MJ with an uncertainty factor (assumed to be the 95% bound) of $10 \text{ MJ}/2.6 \text{ MJ}=3.8$.

Mass and Energy to Heat One MARK-42, Six MURR, or Six MIT Assemblies
in a Storage Location

		MURR	MIT	MK-42
Fuel Meat - Mass	U or PUO ₂	5.0 kg	2.9 kg	3.8 kg
	Al	7.3 kg	1.3 kg	12.9 kg
Clad - Mass	Al	18.1 kg	7.0 kg	10.8 kg
Energy to Heat Meat (565°C)		4.0 MJ	0.85 MJ	6.8 MJ
Energy to Heat Meat + Clad (565°C)		13.0 MJ	4.3 MJ	12.2 MJ

The masses are based on information in N-CLC-H-00136, N-CLC-H-00147, and DPSOP 134 (Ref. 151, 152, 200).

Radionuclide Inventory (INV)

The inventory of isotopes (INV) is determined on a Reference Fuel Assembly (RFA) basis which is fictitious but bounds all isotopic contents of all fuel assembly types. Thus the controlling isotopes would be bounding or conservative for even the worst single fuel assembly that could be melted, and could be very conservative for scenarios in which a significant number of assemblies are melted. It is assumed that the best estimate for the average of several fuel assemblies that might be melted is 1/2 of the RFA inventory value, and take the RFA value as the 95% bound. This is believed very conservative in that the dominant isotopes for offsite dose will be shown below to be Pu and Cm, which are present to a significant extent in only a few of the assemblies. The RFA is dominated by few assembly types, such as the Mark 42. Eliminating the dose-dominant assemblies would drive down the RFA contents significantly for the most significant isotopes of concern.

Release from Fuel (ARF)

The fraction of the isotope released from the fuel (ARF) is taken from the Table 14 of the Restrepo report (Ref. 99) which is also recommended as a source of applicable data by DOE-HDBK-3010-94. Restrepo provides values for oxide fuel only and Mishima does not distinguish between oxide and metallic fuel. Based on the data that is available from the U-Al fuel, which has a much lower melting temperature, no actinide release is usually noted (Ref. 201, 202, and

203). Therefore the use of the Restrepo data is overly conservative for metallic fuel, at least for actinides.

Restrepo also provides the 1-sigma error factors from the data analyzed. These are provided in S-CLC-H-00602 (Ref. 229) and lie mostly in the range of 5 to 20, except for Am and Cm where the 1-sigma error factor is 1.6. For 95% error factors (1.645 sigma) it would be necessary to take these to the 1.645 power which would give exceedingly large uncertainty ranges in many cases. However, on closer examination of the Restrepo data we note that the large 1-sigma error factors, and to some extent even the geometric means or medians, are greatly influenced by data from annealing a small sample for 4 hours to 1750 C (Ref. 99, Appendix D). This is not characteristic of the criticality event in which the water available should quench the fuel in a short time. Therefore, a judgment was made that the Restrepo 1-sigma values for the isotopes other than Am and Cm should already be adequately conservative that they could be considered to represent the 95% EF for the criticality event, as is. The error factors for Am and Cm were, however, raised to the 1.645 power.

Water Leak Path Factor (LPF)

The leak path factor for the water is very significant. The process-induced criticality will most likely occur with a depth of water consistent with the assumptions in Chapter 6 of the RBOF SAR, or approximately 2.1m of overlying water. However, the following discussion is applicable to the minimum water level coverage for fuel in RBOF of 1.37m (4.5 feet), if the uncertainty in the scrubbing factor is recognized, i.e., there is a distribution in the amount of scrubbing as implied by the data and the error factor. One component to the uncertainty in the amount of scrubbing is the water level.

The lower bound scrubbing factors are taken to be 1 for noble gases, 100 for halogens, and 250 for particulate releases (Ref. 99 and 205) at the lower bound is 0.3 m water. The median values for the latter two groups are assumed to be factors of 10 greater, i.e. 1000 and 2500 (~ to 1.65m overlying water depth in the Cunane report (Reference 205)). The median LPF values are the reciprocal of the DF and are 1, 1E-3, and 4E-4 respectively, with an associated error factor of 5. The chalogens, and platinoids, were assumed to have the same release fractions as the halogens. DOE-HDBK-3010-94 (Ref. 75) indicates that under some conditions ruthenium in processing plants can form a gas not very readily removed by water. Discussions with SRS chemists indicate that the volatile ruthenium tetroxide can be formed under acidic oxidizing conditions found in certain separation processes but would not be expected on heating of metallic fuels stored in demineralized water. Therefore this analysis assumes the ruthenium will be scrubbed similarly to halogens.

Dilution Factor (X/Q)

The dilution (X/Q) factor median and 99.5th percentile values for distances of 100m, 640m and the MOI (95th percentile) receptors are tabulated below along with error factor (AXAIRQ

computer model - Appendix A). The 99.5th percentile is used for all but the MOI receptor for this term due to limitations in the AXAIRQ computer model. The use of this statistic rather than the 95th will introduce some conservatism to the overall result at the 100m and 640m.

Dilution Factors (s/m^3) for Various Distances

Distance	50 th Percentile Meteorology	99.5 th percentile Meteorology	Error Factor
100 meters	1.38E-3	3.66E-3	2.65
640 meters	4.42E-5	1.35E-4	3.05
Site Boundary	7.71E-7	3.49E-6*	4.53(95 th)/6.96(99.5 th)

* 95th percentile for MOI case alone.

Consequence Analysis

S-CLC-H-00602 (Ref. 229) provides the full calculational spreadsheet for the onsite receptors at 100m and 640m, respectively. The median doses are given for each isotope, which when summed gives doses of 0.125 rem and 3.98 mrem at the two onsite distances. The bounding dose from compounding all the conservatisms simultaneously by isotope results in a total of 600 rem and 22 rem, respectively. The ratio of the 95th to median doses is the product of all the uncertainty factors, or about 4800 for 100m and approximately 5500 for 640m.

When the uncertainties are convoluted using the 95th percentile methodology, the convoluted uncertainty is about 40 at the two distances. Applying the factor to the median dose at the two distances gives a 95th percentile upper bound dose of 4.9 rem and 0.16 rem, respectively.

In summary, at 100 meters the median dose is 125 mrem and the 95% upper bound dose is 4.9 rem. At 640 meters the median dose is 3.98 mrem and the upper 95% dose is 0.16 rem.

For the dose at the offsite boundary, the only differences in the inputs are the lower median X/Q value and somewhat higher dilution term error factor. For the MOI receptor at 12.1 km, the median dose is 7.0E-05 rem, and 95th percentile dose is 3.3 mrem.

COMPARISON TO GUIDELINES

The risks associated with a process error-induced criticality are shown in Figure 3.4-2 using the methods described in Section 3.4.1. The risks are a result of the consequences and frequencies summarized in Table 3.4-14. The radiological risks to the RBOF associated with a process error-induced criticality are within established EGs as shown in Figure 3.4-2.

SUMMARY OF SAFETY-CLASS SSCs AND TSR CONTROLS

There are two candidate SSCs in the RBOF and RRF. The safety-class function is to retain a -19-inch (below floor level) level of basin water through process-induced criticality events. The two SSCs are the concrete basins and the basin water level indicator (ruler). TSR controls are required to ensure that the minimum basin water level assumed in the analysis is maintained. The basin walls and floor are a passive design feature used for containing the required water level associated with the assumed DF defined above.

3.4.2.4 Severe Weather Accident

The following three severe weather accidents were evaluated:

- Freezing temperatures, which lead to a breach of one of the chemical dumpsters located outside the facility
- High winds, which cause failure of the rain shelter, possibly resulting in a breach of the chemical dumpsters
- High winds, which cause the facility to collapse, with structural debris falling into the pool and disrupting the fuel storage configuration, leading to a criticality and fuel breach

FREEZING TEMPERATURES

Freezing temperatures leading to a breach in one of the chemical dumpster systems was identified as an event to be analyzed in the hazards analysis. The event in the hazards analysis that identified this initiator is Event 348, having moderate consequence and an unlikely frequency, or a risk rank of 5. This event has the same results as the events analyzed in Section 3.4.2.1. The difference is that this scenario only evaluates one initiating event, freezing temperatures, whereas Section 3.4.2.1 evaluates "internal" initiators. Therefore, while the frequency of occurrence is expected to be different, the source term will remain the same as is calculated in Section 3.4.2.1.

Scenario Development

Three chemical dumpsters are located outside of the RBOF building on chemical pads: a 1,000-gallon 50% nitric acid dumpster and a 1,000-gallon 50% sodium hydroxide dumpster, both located on chemical pad No. 1, and a 1,000-gallon 30% sodium nitrite dumpster and associated 1,600-gallon holdup tank located on chemical pad No. 2. The dumpsters are replenished with chemicals by disconnecting the piping and transferring the dumpsters to F Area for filling. The dumpsters are then returned to the RBOF and re-connected.

In this scenario, the chemical dumpsters or hose racks are damaged by heavy snow, hail, or ice storms that result in a containment breach of any one of the three chemical supply systems. The chemical material is then spilled to the bermed pad where it evaporates and disperses into the atmosphere.

Possible Initiating Events

For this scenario, there is only one initiating event, the cold weather-related phenomena.

Accident Event Progression

Upon breach of one of the chemical systems, the material would leak out to the bermed pad where it would either drain into waste tank EP38 or evaporate to the atmosphere.

Preventive Features

The following are preventive features for this scenario:

- Nitric acid dumpster outlet valve (H-22) is normally shut.
- Sodium hydroxide dumpster outlet valve (OH-22) is normally shut.
- Sodium nitrite dumpster outlet valve (SN-3) is normally shut.
- The chemical pads are covered by a protective shelter.

Mitigative Features

The following are mitigative features for this scenario:

- The chemical pads are located in bermed areas that either drain directly to or may be pumped to waste tank EP38. Structural analysis of the chemical pad No. 1 berm has determined that the berm can withstand the impact from a collapsed rain shelter. The forces placed on the berm are well within the berm's capacity without cracking (Ref. 73, 187).
- The low temperatures will cause the sodium nitrite and the sodium hydroxide to solidify upon exposure to the atmosphere. The nitric acid melting point is sufficiently low that it is improbable that it would solidify, although it is possible that the viscosity of the material would increase and result in a lower rate of evaporation.
- The low temperatures under which the spill occurs are likely to inhibit evaporation of any of the chemicals. The meteorological conditions (e.g., clouds, snow) will also contribute to lower rates of evaporation.

Frequency Binning

The frequency determination for a severe weather-initiated chemical spill at the RBOF and RRF is based upon SRS climatological data and upon conservative conditional probabilities for ensuing events required for the initiator to result in a large chemical spill. The frequency is estimated as $6.94\text{E-}04/\text{yr}$, which is in the unlikely ($10^{-2}/\text{yr} > p > 10^{-4}/\text{yr}$) frequency bin. The analysis is documented in S-CLC-H-00281 (Ref. 179) and is summarized as follows.

The severe weather event consists of freeze damage due to cold temperatures and structural damage due to heavy snow, hail, or ice storms. According to the hazards analysis, freezing of the dumpsters and associated piping is postulated to be a remote possibility of negligible consequence. Therefore, damage due to freezing temperatures is not considered, and the evaluation focuses on the possibility of structural damage to the dumpster systems and the hose rack.

The sodium hydroxide and nitric acid dumpsters are installed on chemical pad No. 1. No other dumpsters are installed on this pad. The sodium nitrite dumpster and the associated holdup tank are installed within chemical pad No. 2. One other tank is also installed on this pad and is no longer in service. Chemical pad No. 1 is located outside and immediately to the north of the RRF. Chemical pad No. 2 is located outside and to the west of the RRF. Hose racks are located inside each hose box. Hose box No. 1 is located immediately to the west of the RRF. Hose box No. 2 is located on chemical pad No. 2.

The potential for accumulation of heavy snow, hail, or ice on the exterior surfaces of the dumpster systems and the hose boxes is not a concern relative to the structural integrity of the components. Also, the accumulation of wind-blown precipitation is considered to be inconsequential because the dumpsters and the hose rack will be mostly enclosed within the steel structure.

This analysis considers the structural integrity of the protective steel shelter over the chemical pads. According to Chapter 2, the steel shelter is designed to conform to Uniform Building Code General Service construction. The roof is designed to withstand a load of at least 20 lb/ft^2 in addition to dead weight (Ref. 193).

According to Table 1 in WSRC-RP-89-313 (Ref. 194), the maximum monthly precipitation of snow, hail, and ice pellets in a 36-year period in the Augusta, Georgia, area is 14 inches in 1973. According to Chapter 6 of Marks Standard Handbook for Mechanical Engineers (Ref. 183), the average density of ice is 56 lb/ft^3 , and the average density of snow is 8 lb/ft^3 . Therefore, 14 inches of snow accumulation would equate to 9.33 lb/ft^2 , and 14 inches of ice accumulation would equate to 65.33 lb/ft^2 . Thus, it is assumed that this event has the potential to challenge the structural integrity of the shelter if the precipitation actually accumulates on the shelter and if a significant portion of the accumulation is ice.

The initiating event frequency is taken to be one occurrence in 36 years (or 0.028 occurrences/yr). Since the steel structure will be constructed with a sloping roof for runoff of

precipitation, it is considered unlikely that there will be an accumulation on the order required to exceed the design capacity of the roof during a 14-inch accumulation event. Accordingly, a conditional probability of 0.1 (Ref. 91) is multiplied by the initiator to account for this effect. A conditional probability of 0.5 (Ref. 91) is assumed for the accumulation to consist significantly of ice. Furthermore, a probability of 0.5 (Ref. 91) is assumed that the structural failure of the shelter actually results in damage to the dumpster system or hose rack sufficient to cause a large chemical spill.

The likelihood of damage to the dumpsters and the hose boxes due to collapse of the steel shelter is calculated as the product of the frequency of occurrence of a 14-inch accumulation of snow, hail, and ice pellets and the assumed probabilities as follows:

$$(0.028 \text{ occurrences/yr})(0.1)(0.5)(0.5) = 6.94\text{E-}04/\text{yr}$$

No credit is taken for the ability of the diked area surrounding each chemical pad to contain and collect a chemical spill from any of the dumpsters. No credit is taken for the drain line in chemical pad No. 1 and for the sump and associated pump in chemical pad No. 2 to direct any chemical spill to waste tank EP38. Chemical pad No. 2 is designed to hold at least one dumpster full of sodium nitrite; however, it cannot contain the contents of both the dumpster and hold tank.

Source Term Analysis

The source terms for the three possible chemical spills (50% nitric acid, 50% sodium hydroxide, and 30% sodium nitrite) are the same as they are reported in Section 3.4.2.1. The release under the freezing conditions is expected to yield lower source terms due to the conditions outlined under the heading "Mitigative Features" above. Therefore, the scenario used to calculate the source terms in Section 3.4.2.1 is considered bounding for this low temperature scenario.

Section 3.4.2.1 indicates that sodium hydroxide will be disbursed at a rate of $3.5\text{E-}06$ kg/sec, nitric acid will be disbursed at a rate of 0.0673 kg/sec, and sodium nitrite will be disbursed at a rate of $4\text{E-}06$ kg/sec.

Consequence Analysis

The consequences are the same as those presented in Section 3.4.2.1. For the sodium nitrite and sodium hydroxide releases, the consequences of the release are insignificant and there is no need for further analysis at greater distances. The consequences for the receptors located at the OEP and at the site boundary are acceptable. (Refer to the hazardous chemical worker safety section of Section 3.3.2.3 for the onsite receptor consequence methodology and analysis.)

Comparison to Guidelines

Based on the frequency for this event, the comparison to EGs is based on the analysis in Section 3.4.2.1.

Summary of Safety-Class SSCs and TSR Controls

This accident scenario does not challenge the offsite EGs; therefore, no safety-class SSCs are required. (Refer to the hazardous chemical worker safety section of Section 3.3.2.3 for the onsite receptor consequence methodology and analysis.)

WIND-INDUCED RAIN SHELTER FAILURE

High winds leading to failure of the rain shelter and breach of the chemical dumpsters was identified as an event to be analyzed in the hazards analysis. The event in the hazards analysis that identified this initiator is Event 347, having moderate consequence and an unlikely frequency, or a risk rank of 5. This event has the same results as the events analyzed in Section 3.4.2.1. The difference is that this scenario only evaluates one initiating event, wind-induced rain shelter failure, whereas Section 3.4.2.1 evaluates "internal" initiators. Therefore, while the frequency of occurrence is expected to be different, the source term will remain the same as is calculated in Section 3.4.2.1.

Scenario Development

Three chemical dumpsters are located outside of the RBOF building on chemical pads: a 1,000-gallon 50% nitric acid dumpster and a 1,000-gallon 50% sodium hydroxide dumpster, both located on chemical pad No. 1, and a 1,000-gallon 30% sodium nitrite dumpster and associated 1,600-gallon holdup tank located on chemical pad No. 2. The dumpsters are replenished with chemicals by disconnecting the piping and transferring the dumpsters to F Area for filling. The dumpsters are then returned to the RBOF and re-connected.

In this scenario, the chemical dumpsters or hose racks are damaged by the collapsed rain shelter that results in a containment breach of any one of the three chemical supply systems. The chemical material is then spilled to the bermed pad where it evaporates and disperses into the atmosphere.

Possible Initiating Events

For this scenario, there is only one initiating event, high winds.

Accident Event Progression

Upon breach of one of the chemical systems, the material would leak out to the bermed pad where it would either drain into waste tank EP38 or evaporate to the atmosphere.

Preventive Features

The following are preventive features for this scenario:

- Nitric acid dumpster outlet valve (H-22) is normally shut.
- Sodium hydroxide dumpster outlet valve (OH-22) is normally shut.
- Sodium nitrite dumpster outlet valve (SN-3) is normally shut.

Mitigative Features

The following is a mitigative feature for this scenario:

- The chemical dumpsters are located in bermed areas that slope to the waste tank EP38 drain, which reduces the amount of chemicals that will evaporate.
- The chemical pad is covered by a rain shelter. The roof does protect part of the pad from direct sunlight, thus reducing the rate of evaporation of spilled chemicals.
- The chemical pads are located in bermed areas that either drain directly to or may be pumped to waste tank EP38. Structural analysis of the chemical pad 2 berm has determined that the berm can withstand the impact from a collapsed rain shelter. The forces placed on the berm are well within the berm's capacity without cracking (Ref. 73, 187). (Credited).

Frequency Binning

For the initiating event of rain shelter collapse, the rain shelter has been structurally evaluated against the wind loads associated with a PC-2 wind speed or 84 mph with an AEP of 1E-02/yr. On the basis of the structural analysis documented in T-CLC-H-00244 (Ref. 117), the frequency of the rain shelter collapsing due to wind speeds less than or equal to 84 mph is 2.70E-02/yr (Ref. 195). Straight wind initiators having an AEP of less than 1E-02/yr are considered beyond evaluation basis and are not analyzed.

Source Term Analysis

The source terms for the three possible chemical spills (50% nitric acid, 50% sodium hydroxide, and 30% sodium nitrite) are the same as they are reported in Section 3.4.2.1. The release under

the freezing conditions is expected to yield lower source terms due to the conditions outlined under the heading "Mitigative Features" above. Therefore, the scenario used to calculate the source terms in Section 3.4.2.1 is considered bounding for this low temperature scenario.

Section 3.4.2.1 indicates that sodium hydroxide will be disbursed at a rate of $3.5\text{E-}06$ kg/sec, nitric acid will be disbursed at a rate of 0.0673 kg/sec, and sodium nitrite will be disbursed at a rate of $4\text{E-}06$ kg/sec.

Consequence Analysis

The consequences are the same as those presented in Section 3.4.2.1. For the sodium nitrite and sodium hydroxide releases, the consequences of the release are insignificant and there is no need for further analysis at greater distances. The consequences for the receptors located at the OEP and at the site boundary are acceptable. (Refer to the hazardous chemical worker safety section of Section 3.3.2.3 for the onsite receptor consequence methodology and analysis.)

Comparison to Guidelines

Based on the frequency for this event, the comparison to EGs is based on the analysis in Section 3.4.2.1.

Summary of Safety-Class SSCs and TSR Controls

This accident scenario does not challenge the offsite EGs; therefore, no safety-class SSCs are required. (Refer to the hazardous chemical worker safety section of Section 3.3.2.3 for the onsite receptor consequence methodology and analysis).

HIGH WIND-INDUCED CRITICALITY

The hazards analysis identified Events 63, 63a and 64 as high wind-induced criticality events. Event 63 involves the deformation of fuel storage racks due to the effects of the facility structure or crane damage impacting the storage array, with the resulting fuel disruption leading to a criticality event. Event 63a involves the same scenario as event 63, but with the addition of a basin draindown.. Event 64 involves fuel collecting in an unsafe geometry from a roof collapse.

Scenario Development

The potential exists for high winds to cause the roof structure and possibly the entire facility to fail, causing structural debris to fall into the storage basin.

Possible Initiating Events

For this scenario, the initiating event is high winds present during a hurricane (straight winds) or tornado.

Accident Event Progression

A 119-mph wind on the girts and siding will result in a postulated failure of the high bay girts. Due to pressure change, the windward siding and girts will fall inwards. The critical wall then becomes the 8-inch Concrete Masonry Unit (CMU) wall on the 4.8 line. A 136-mph wind will cause a collapse of the office building extension due to the loss of its wall diaphragms. The wind will then be directed on the 12-inch exterior CMU walls, causing them to become breached, which will result in the collapse of the low roof and cause damage to the unmodified 12-inch 6-line wall. A 191-mph wind will cause a failure of the vertical bracing on the G and E lines. This failure will cause the steel structure to collapse. At the same time, the 12-inch exterior CMU wall, the 12-inch 6-line wall, the 12-inch D-line wall, and the 8-inch wall on the 4.8 line will fail. The criticality event is assumed to result in a $5E+18$ fission yield, as discussed in Section 3.4.1.5.

Preventive Features

Measures to prevent a high wind-induced criticality at the RBOF include the following:

- The original design is based on a wind speed of 62 mph and includes a uniform 20-lb/ft² loading to account for wind effects.
- Storage racks are configured and spaced to prevent criticality events. (Credited)
- NSDSs specify safe spacing distances between fuel bundles. (Credited)

Mitigative Features

The following are mitigative features for this scenario:

- Fuels are stored in water-filled pools under a minimum of 4-1/2 feet of water for scrubbing of fission products and attenuation of direct dose from criticality. (Credited)
- The NIMs alerts operators that a criticality event with a fission product release is in progress.
- Emergency response plans and onsite medical treatment personnel facilitate removal and treatment of injured workers.
- Site audible warning systems alert workers of the need to evacuate the area.

- Workers are trained to respond safely in the event of a criticality or high wind event. (Credited)

Frequency Binning

The frequency determination for high wind-induced criticality at the RBOF is based upon high wind hazard curves for SRS, fragility estimates for structural components at risk, and conditional probabilities of resulting criticality. The frequency assessment considers both straight wind and tornado. The magnitude of the initiators is based on criteria established for the PC of the facility. This magnitude establishes the "evaluation basis" that the facility is analyzed against. The frequency of criticality due to an evaluation basis straight wind initiator is estimated as $3.93\text{E-}05/\text{yr}$, and the frequency of criticality due to an evaluation basis tornado initiator is estimated as $7.87\text{E-}07/\text{yr}$. The overall frequency of an evaluation basis high wind-induced criticality is the sum of the frequency of an evaluation basis straight wind-induced criticality and the frequency of an evaluation basis-induced tornado criticality (Events 63, 63a and 64), or $4.01\text{E-}05/\text{yr}$, which is in the extremely unlikely ($10^{-4}/\text{yr} > p > 10^{-6}/\text{yr}$) frequency bin. The analysis is documented in S-CLC-H-00340 (Ref. 188) and is summarized as follows.

The straight wind event and the tornado event analyzed are based upon criteria for a PC-3 facility. PC-3 criteria establish that the facility withstand a straight wind event with an AEP of $1.0\text{E-}03$, and a tornado event with an AEP of $2.0\text{E-}05$. While this may or may not be consistent with the original design basis for the facility, it establishes an evaluation basis against which the facility must be analyzed. Therefore, straight wind initiators having an AEP $\geq 1.0\text{E-}03$, and tornado initiators having an AEP $\geq 2.0\text{E-}05$, are considered in the determination of criticality frequency, whereas straight wind initiators having an AEP $< 1.0\text{E-}03$, and tornado initiators having an AEP $< 2.0\text{E-}05$, are considered beyond design/evaluation basis. Based on the hazard curve used in this analysis, a straight wind event with an AEP of $1.0\text{E-}03$ corresponds to a speed of approximately 110 mph, and a tornado event with an AEP of $2.0\text{E-}05$ corresponds to a speed of approximately 155 mph.

During the initial phases of analysis, the Structural Mechanics Section of the Projects, Engineering, and Construction Division (PECD) provided fragility values for two particular components at risk due to a wind event: CMU walls and the entire building structure. The CMU walls were identified separate from the structure because they were believed to be the weakest part of the facility (non-reinforced and non-load bearing). On the basis of the identified vulnerabilities, modifications were made to the RBOF facility (Ref. 196) to increase the wind resistance of the steel superstructure to meet the PC-3 wind speeds. Following the modifications, the resistance of the CMU walls increased such that their failure was no longer governing. In addition, failure of the steel superstructure was found to not significantly impact the integrity of the RBOF basin and the potential for basin draindown. Refer to Section 3.4.2.5 for discussion of a breach of pool basin due to seismic initiators.

An event tree for evaluation basis straight wind-induced criticality was developed and is shown in Figure 3.4-3. A similar event tree for evaluation basis tornado-induced criticality was

developed and is shown in Figure 3.4-4. Therefore, the event trees given in Figures 3.4-3 and 3.4-4 consider whether or not the structure collapses during an evaluation basis high wind event (top event NO-STRUCT) and whether or not a criticality results given structure failure (top event NO-CRIT-STRUCT).

EXCEL spreadsheets were constructed to calculate the event tree sequence frequencies. The spreadsheets are given in S-CLC-H-00340. The sequence of concern is SEQ-3W for evaluation basis straight wind-induced criticality, and SEQ-3T for evaluation basis tornado-induced criticality. The spreadsheets are set up to calculate the frequency of the wind event, and the probability of success or failure of each event tree branch point for straight wind speeds from 10 mph to approximately 110 mph in 0.2-mph increments and for tornado speeds from 50 mph to approximately 155 mph in 0.2-mph increments. The AEP of the various wind events is based on the high wind hazard curve for SRS as given in UCRL-53526 (Ref. 115). Two equations were derived for this curve using a third order polynomial fit of the data. These equations are as follows:

For straight winds,

$$AEP = 10^{\left(1.3431 - 3.8994 \times 10^{-2} * WS - 1.9754 \times 10^{-7} * WS^2 + 1.8068 \times 10^{-9} * WS^3\right)} \quad (\text{Eq. 3.4-27})$$

For tornadoes,

$$AEP = 10^{\left(1.9755 - 1.7397 \times 10^{-2} * WS - 1.4656 \times 10^{-6} * WS^2 + 2.2863 \times 10^{-9} * WS^3\right)} \quad (\text{Eq. 3.4-28})$$

The tornado wind speeds must be converted to equivalent fastest-mile wind speeds to determine structural responses. The formula for this conversion from UCRL-15910 (Ref. 36) is as follows:

$$V_{fm} = 0.958 * V_t - 11.34 \quad (\text{Eq. 3.4-29})$$

where:

V_{fm} = fastest mile wind speed

V_t = tornado speed

The probability of building structural failure is based on fragility estimates for these components, and the following formula from EPD-SE-0115 (Ref. 197):

$$P = \Phi \left(\frac{\ln \left(\frac{WS}{MC} \right)}{\sqrt{\beta_R^2 + \beta_u^2}} \right) \quad (\text{Eq. 3.4-30})$$

where:

P	=	probability of failure
ϕ	=	standard normal cumulative distribution function
WS	=	wind speed
MC	=	median capacity of the structural component
β_R	=	random uncertainty parameter
β_U	=	modeling uncertainty parameter

The median capacity and the uncertainty parameters for the building structure are provided below (Ref. 117). The CMU walls are not the governing failure as discussed in Chapter 2.

Structural Component	Median Capacity (MC)	Random Uncertainty Parameter (β_R)	Modeling Uncertainty Parameter (β_U)
Steel Superstructure	191 mph	0.28	0.20

The probability of criticality given building structure failure was developed based on expert opinion. The opinion elicitation and selection process is documented in S-CLC-H-00350 (Ref. 150) and resulted in the following conditional probability:

Probability of criticality given building structure failure = 0.1

This probability was developed consistent with rules for developing split fractions for an Accident Progression Event Tree (Ref. 91).

The overall frequencies for sequences in which a criticality occurs during an evaluation basis straight wind event (sequence SEQ-3W), and the overall frequencies for sequences in which a criticality occurs during an evaluation basis tornado event (sequence SEQ-3T) are obtained by summing the frequencies calculated for each incremental wind speed in the EXCEL spreadsheets. The frequency for SEQ-3W is 3.93E-05/yr. The frequency for SEQ-3T is 7.87E-07/yr.

Of additional interest to the consideration of NPH-induced criticality is the postulated scenario of fuel rack failure following a basin draindown induced by failure of the filter-deionizer system. This scenario was identified as the only potentially credible scenario that would challenge basin water level assumptions and associated DFs used in the source term release calculations. In the postulated scenario, a failure of the filter-deionizer system occurs during a 112-hour facility shutdown, resulting in a basin water level reduction. Before the water level deficiency is detected and corrected, an earthquake or high wind event occurs.

The frequency of rack failure due to evaluation basis natural phenomena events are calculated by adding the sequence frequencies involving rack failure. SEQ-3W (for high wind events) and SEQ-3T (for tornadoes) involve rack failure. Seismic sequences involving rack failure include SEQ-3S, SEQ-4S, SEQ-5S, SEQ-6S, SEQ-11S, SEQ-12S, SEQ-13S, and SEQ-14S. The total frequency of evaluation basis NPH-induced rack failure is as follows:

SEQ-3W	3.93E-05/yr
SEQ-3T	7.87E-07/yr
SEQ-3S	2.83E-09/yr
SEQ-4S	7.47E-15/yr
SEQ-5S	2.55E-08/yr
SEQ-6S	6.73E-14/yr
SEQ-11S	3.73E-17/yr
SEQ-12S	1.51E-22/yr
SEQ-13S	3.77E-16/yr
SEQ-14S	<u>1.53E-21/yr</u>
	4.01E-05/yr

A search of the SRS fault tree databank indicates that there has been one recorded incident of a filter-deionizer leak in 32 years of RBOF operation. This value equates to an initiation frequency of 3.10E-02/yr.

The combined frequency of filter-deionizer leakage followed by an evaluation basis NPH-induced rack failure during a 112-hour shutdown is 1.59E-08/yr.

It can be concluded that this scenario is beyond extremely unlikely and that related water level assumptions are not challenged.

Wind-Induced Criticality Source Term Analysis

A criticality will produce fission products and energy. A portion of this energy will be deposited in the fuel in the form of heat. It is this heat that can lead to the melting of the fuel and clad. Heat is the mechanism both for creating a leak path for the source term (clad melting) and for volatilizing a portion of the fuel (fuel melting). The escaping volatile and nonvolatile isotopes represent the source term for the criticality.

The MAR from a 1.0E+20 fission yield was developed in N-CLC-H-00138 (Ref. 190) as shown in Table 3.4-5. That MAR is linearly scaled down to a 5.0E+18 fission yield and is presented in Table 3.4-6. DRs, ARFs, and LPFs were then applied to the MAR to determine the source term.

The source term for this event is fully detailed in S-CLC-H-00399 (Ref. 191). The source term is represented by Equation 3.4-25.

Material at Risk

The MAR for the criticality contains both radionuclides originally in the fuel and those created during the criticality. In an earlier work, the fission products for 10^{20} fissions were calculated using ORIGEN (Ref. 190). This unscaled fission product MAR is given in Table 3.4-5. These values must be scaled by a factor of 0.05 for the $5\text{E}+18$ fissions expected (Table 3.4-6).

The MAR for the radionuclides originally in the fuel is bounded by the number of RFAs that could possibly melt during the event. The RFA is a hypothetical bundle (assembly) that was developed to represent a bundle that would produce a bounding consequence for RBOF accident analysis (Ref. 21). To determine the number of involved assemblies, a conservative analysis was performed, which assumed that all of the released fission energy would go towards adiabatically heating of the fuel (Ref. 191). It is this heat that leads to the melting of the fuel cladding and fuel meat. Heat is the mechanism both for creating a leak path for the source term (clad melting) and for volatilizing a portion of the fuel (fuel melting).

Using the fuel and cladding characteristics of the MURR and MIT fuels and the conservative inventory of RFA, the calculation estimates that 37.1 RFAs will melt during a $5\text{E}+18$ fission yield event. Therefore, the amount of isotopes available for release from the melted fuel is determined by the inventory of an RFA multiplied by 37.1 (Table 3.4-7).

Adjustment Factors to the Source Term

In a criticality event, two factors will mitigate the release. The first factor is the ARF, which is the amount of material that will escape from the damaged fuel into the water. The second factor is the leak path factor, which is the percentage of isotopes that will escape from the water and become airborne. No credit is taken for the building exhaust filters or the exhaust ventilation system.

In DOE-HDBK-3010-94 (Ref. 75), Mishima suggests a value of 0.1 for the DR for radionuclides generated by a criticality involving metal pieces. For the purposes of this calculation, the DR will be assumed to be 1. The ARFs used come directly from Table 6-10 of DOE-HDBK-3010-94.

The LPFs specified in DOE-HDBK-3010-94 (Ref. 75) define the amount and type of isotopes expected to escape the basin water (once released from the melted fuel) and become airborne from the basin water surface. The values associated with the noble gases and halogens (Table 3.4-6) are not dependent upon water depth; consequently, the LPF is specified as 1.0, or 100% of the released isotopes from the melted fuel become airborne. For the remaining groups of chemical classes, the amount of isotopes released is strongly dependent upon several factors,

including the amount of water covering the fuel which is assumed to melt, the amount of basin water subcooling, particulate size, and the carrier gas-steam mass fraction (Ref. 99).

According to Reference 99, DFs (or 1/LPF) on the order of 25 to 30 are achieved simply due to the pool being subcooled. Combined with a pool depth of 6 meters (19 feet), this could lead to DFs for particulates on the order of 20,000. For the RBOF analysis, a DF of 2,000 ($1/5E-04$) is conservatively assumed since the height at which the straight wind-induced criticality event would occur would be at or near the bottom of the basin due to the building failure. The DF assumed is consistent with the previous minimum basin levels associated with -19 inches below the floor level defined in the past safety analyses (Ref. 46). Thus, this reduction in the assumed DF also accounts for basin water subcooling. Controls currently exist for maintaining the basin water temperatures at or below 40°C. Subcooling is defined in Reference 99 as 27°C, which for the RBOF may be exceeded at times. Therefore, the reduction in DF from a suggested value of 20,000 to an assumed value of 2,000 would account for any conditions where subcooling cannot be ensured.

The ARF and LPF values are taken from DOE-HDBK-3010-94 and are presented in Table 3.4-6. The overall reduction factors appear in Table 3.4-2.

Source Term

The source term associated with the inventory in the basin (fuel melting) is given in Table 3.4-7. The total source term is then arrived at by adding to the product of the fission products (Table 3.4-5), fission scale factor (0.05), and the overall release fractions (Table 3.4-2).

Consequence Analysis

Radiological consequences resulting from an NPH event (earthquake or high winds) are calculated for the MOI and the individual at the OEP using the methods described in Section 3.4.1 and are summarized in Table 3.4-14. The NPH event is modeled to produce several events and source terms as described above. Results are substantiated in S-CLC-H-00405 (Ref. 192) for the criticality and fuel melt, S-CLC-H-00570 (Ref. 198) for the fuel breach, and S-CLC-H-00428 (Ref. 199) for the waste tank breach.

As an alternative to the bounding consequence analysis performed above, the following analysis is provided to illustrate the methodology (Section 3.4.1.6) for propagating uncertainties in a dose calculation. The scenario is a criticality event caused by high straight winds, which cause the heavy building superstructure to fall into the basin, distorting storage racks and fuel to the extent that a critical fuel geometry is reached. The information presented in this section is contained in S-CLC-H-00426 (Ref. 177). The resulting doses from this alternative methodology and analysis will be used for comparison to the established EGs in lieu of the bounding doses previously established.

Dose Calculation

The dose is estimated from the following equation and summed over the significant isotopes:

$$\text{Dose} = \text{FIS} * \text{EGY} * (1/\text{EMELT}) * \text{INV} * \text{ARF} * \text{LPF} * (\chi/Q) * \text{B} * \text{DCF} \quad (\text{Eq. 3.4-31})$$

where:

Dose	=	inhalation dose (rem effective dose equivalent)
FIS	=	number of fissions
EGY	=	3.28E-17 MJ/fission
EMELT	=	energy to melt a fuel assembly (MJ)
INV	=	inventory of the isotope in an assembly (Ci)
ARF	=	fraction of the isotope released from the fuel on melting
LPF	=	fraction of isotope not removed by the water cover
χ/Q	=	dispersion factor to site boundary (sec/m ³)
B	=	standard man breathing rate of 1.25 m ³ /hr (=3.47E-04 sec/m ³)
DCF	=	dose conversion factor (rem per Ci of the isotope inhaled)

Parameter Values and Uncertainty Range

Several of these parameters have substantial uncertainty as follows.

Fission Burst (FIS)

The number of fissions in criticality events with water-moderated solid fuel varies substantially. A log-normal distribution fit of the data (see EPD-CTG-950067 and S-CLC-H-00343 [Ref. 77, 175]) indicates a median of FIS=1.2E+17 fissions with a 95% uncertainty factor of 30 for criticalities that could be characteristic of NPH damage. (In energy units, this value corresponds to a 50%, or median, fission burst energy of 3.9 MJ and a 95% upper bound of 118 MJ.)

Energy to Melt Assembly (EMELT)

Metallic (U-Al) type fuel must be heated up into the 600°C range before melting and releasing radioisotopes and must be heated to even to higher temperatures for oxide fuels. The fission energy required to melt a fuel assembly also depends on how fast the energy is generated in comparison with heat transfer from the fuel matrix to the clad and coolant. A best estimate is the energy to heat both the fuel and clad (565°C). From the table below (which shows the energy to heat the fuel in one storage slot for three fuel types), this energy would average about 10 MJ. In

a slow transient, this energy would be even higher due to heat loss to the surrounding water. Furthermore, since the energy generation in the fuel involved in the criticality will not be uniform, some of the fuel will be heated more than 565°C and some less than 565°C, which would decrease the number of assemblies that can be heated to 565°C with a given heat input.

The maximum number of assemblies that could melt from a given energy deposition is that with the lowest energy to melt, which for the MIT fuel (conservatively low-heat-capacity fuel) is calculated to be 4.3 MJ (See table of masses below, which is based on information in References 151, 152, and 200 and the assumption that the storage location would accommodate six MURR or MIT assemblies). A lower energy of 0.85 MJ per assembly could theoretically result if it were assumed that all the energy remains in the fuel meat. However, because of the very thin (0.01 to 0.02 inch) fuel layer thickness in this fuel having a fuel to clad thermal time constant on the order of 1 msec, this is not possible even in a very rapid transient (3 to 5 msec periods) similar to those achieved in the SPERT super prompt critical burst tests.

The lower bound is assumed to be heating the fuel meat and inner half of the clad thickness, which is calculated to require a total of 2.6 MJ per assembly for the MIT fuel (most limiting) (Note that for the MARK-42s, which have a much longer fuel to clad time constant, the energy to heat only the fuel meat is significantly greater than this). Thus, the median value for EMELT=10 MJ with an uncertainty factor (assumed to be the 95% bound) of 10 MJ/2.6 MJ=3.8.

Mass and Energy to Heat One MARK-42, Six MURR, or Six MIT Assemblies
in a Storage Location

		MURR	MIT	MK-42
Fuel Meat - Mass	U or PUO ₂	5.0 kg	2.9 kg	3.8 kg
	Al	7.3 kg	1.3 kg	12.9 kg
Clad - Mass	Al	18.1 kg	7.0 kg	10.8 kg
Energy to Heat Meat (565°C)		4.0 MJ	0.85 MJ	6.8 MJ
Energy to Heat Meat + Clad (565°C)		13.0 MJ	4.3 MJ	12.2 MJ

The masses are based on information in N-CLC-H-00136, N-CLC-H-00147, and DPSOP 134 (Ref. 151, 152, 200).

Radionuclide Inventory (INV)

The inventory of isotopes (INV) is determined on an RFA basis, which is hypothetical but bounds all isotope contents of all fuel assembly types. Thus, the controlling isotopes would be bounding or conservative for even the worst single fuel assembly that could be melted and could

be very conservative for scenarios in which a significant number of assemblies are melted. It is assumed that the best estimate for the average of several fuel assemblies that might be melted is 1/2 of the RFA inventory value and take the RFA value as the 95% bound. This assumption is believed to be very conservative in that the dominant isotopes for offsite dose will be shown below to be plutonium and curium, which are present to a significant extent in only a few of the assemblies.

Release from Fuel (ARF)

The fraction of the isotope released from the fuel (ARF) is taken from Table 14 of the Restrepo report (Ref. 99). Restrepo provides values for oxide fuel only, and DOE-HDBK-3010-94 (Ref. 75) does not distinguish between oxide and metallic fuel. Based on the data that are available from the U-Al fuel, which has a much lower melting temperature, no actinide release is usually noted (Ref. 201, 202, 203). Therefore, the use of the Restrepo data is overly conservative for metallic fuel, at least for actinides.

Restrepo also provides the 1-sigma error factors from the data analyzed. These factors lie mostly in the range of 5 to 20, except for americium and curium where the 1-sigma error factor is 1.6. For 95% error factors (1.645 sigma), it would be necessary to take these to the 1.645 power, which would give exceedingly large uncertainty ranges in many cases. However, on closer examination of the Restrepo data, it is noted that the large 1-sigma error factors, and to some extent even the geometric means or medians, are greatly influenced by data from annealing a small sample for 4 hours to 1,750°C (Ref. 99, Appendix D). These data are not characteristic of the criticality event in which the water available should quench the fuel in a short time. Therefore, a judgment was made that the Restrepo 1-sigma values for isotopes other than americium and curium should already be adequately conservative and that they could be considered to represent the 95% error factor for the criticality event, as is.

Water Leak Path Factor (LPF)

The LPF for the water is very significant. The NPH-caused criticality is assumed to occur in the lower regions of the basin. The lower bound scrubbing factors with 5 or 6 meters of water available are taken to be 1 for noble gases, 100 for halogens, and 1,400 for particulate releases (Ref. 99, 204, 205). The median values for the latter two groups are assumed to be factors of 10 greater (i.e., 1,000 and 14,000). The median LPF values are the reciprocal of the DF and are 1, 1E-03, and 7E-05, respectively, with an associated error factor of 10. The platinoids were assumed to have the same release fractions as the halogens.

In DOE-HDBK-3010-94 (Ref. 75), Mishima indicates that under some conditions ruthenium in processing plants can form a gas not very readily removed by water. Discussions with SRS chemists indicate that the volatile ruthenium tetraoxide can be formed under acidic oxidizing conditions found in certain separation processes but would not be expected upon heating of

metallic fuels stored in demineralized water. Therefore, this analysis assumes that the ruthenium will be scrubbed similarly to halogens and particulates.

Dispersion Factor (χ/Q)

The meteorology (χ/Q) factor median (50%) and upper 95% values for the site boundary, 640-meter distance, and 100-meter distance were obtained from S-CLC-H-00405 (Ref. 192) and are tabulated below along with the 95%/50% error factor.

Dispersion Factors (sec/m^3) for Various Distances

Distance	50% Meteorology	95% Meteorology	Error Factor
Site boundary	7.42E-07	3.53E-06	4.76
640 meters	4.41E-05	1.19E-04	2.70
100 meters	1.35E-03	3.17E-03	2.35

Consequence Analysis

S-CLC-H-00426 provides the median dose by isotope and the total (0.013 mrem). The bounding dose from compounding all the conservatisms simultaneously by isotope gives a total of 1.35 rem. The ratio of these doses is the product of all the uncertainty factors, which is seen to be very large, a factor of over 100,000.

By contrast, the uncertainties are convoluted using the above methodology arriving at an overall upper 95% uncertainty, which yields a factor of uncertainty of 167 for this case. Applying this methodology to the median dose of 0.013 mrem gives a 95% bound of 2.2 mrem. (Note that this is a factor of about 600 below the value when all six factors are compounded simultaneously.)

It is also noted that the dominant dose contribution is from plutonium and curium, followed by ruthenium and strontium. As the recommended methodology suggests, when the dose is not completely dominated by one isotope and those contributing have different release fractions, the pseudo error factors for these isotopes are determined based on the sensitivity of the total dose to the variation of that factor from the median as follows:

$$\text{EF}(\text{pseudo}) = 1 + (\text{EF} - 1) * (\text{isotope's median dose} / \text{total median dose}) \quad (\text{Eq. 3.4-32})$$

This equation was used in determining the effective (or pseudo) error factors for the above isotopes because of the significantly different uncertainty factors for their ARFs.

Comparison to Guidelines

The risks associated with an NPH event are shown in Figure 3.4-2 using the methods described in Section 3.4.1. The risks are a result of the consequences and frequencies summarized in Table 3.4-14. The radiological risks to the RBOF associated with an NPH event are within established EGs as shown in Figure 3.4-2.

Summary of Safety-Class SSCs and TSR Controls

There are two candidate SSCs in the RBOF and RRF. The safety-class function is to maintain a basin water level no lower than -19 inches (below floor level) through NPH-induced events. The two SSCs are the concrete basins and the basin water level indicator (ruler). TSR controls may be required to ensure that the minimum basin water level assumed in the analysis is maintained. The basin walls and floor are a passive design feature used for containing the required water level associated with the assumed DF defined above. Passive design features require no specific TSR LCO controls.

HIGH WIND-INDUCED FUEL BREACH

The hazards analysis identified the possibility of a fuel breach as the result of high winds causing the facility to fail and the debris to fall into the storage basin and breach fuel bundles and /or assemblies. The hazards analysis event number is Event 18.

Scenario Development

In this scenario, the breached fuel would release the fission products contained within the fuel matrix, causing an airborne radionuclide release.

Possible Initiating Events

For this scenario, only one initiating event is examined, high winds.

Accident Event Progression

Assuming that evaluation basis high winds cause the facility to fail, the roof structure and possibly the entire facility could fail, causing roof sections and/or the overhead crane to collapse into the storage basin and breached fuel elements. The fission products contained within the fuel matrix would then be released to the environment.

Preventive Features

Preventive features include the following:

- Storage racks will distribute the weight of the debris.
- Fuel storage containers are structurally sound.
- Basin water will reduce the momentum of the falling debris.

Mitigative Features

Mitigative features include the following:

- The required aging of fuel is met prior to receipt at the RBOF. (Credited)
- Basin water will scrub out some of the released material. (Credited)

Frequency Binning

The frequency at which the fuel could breach due to falling debris is assumed to occur at the frequency calculated for the high wind-induced building failure. This frequency is $3.93\text{E-}04/\text{yr}$ an order of magnitude higher than the high wind-induced criticality frequency of $3.93\text{E-}05/\text{yr}$.

Source Term Analysis

The source term is the amount of radioactive material (in grams or curies) released to the air as the result of a postulated accident. This source term is typically estimated using Equation 3.4-25.

Material at Risk

As stated above, the postulated accident results in physical damage of the cladding of individual fuel assemblies in the RBOF/RRF storage basin.

A worst-case RFA has been defined for the RBOF that reflects the bounding radionuclide inventory of any fuel assembly that is present in, or planned for shipment to, the RBOF (Ref. 21). The RFA is assumed to be constructed of a pellet type fuel matrix with fuel rod cladding. This construction allows the fraction of fission products that have migrated from the fuel matrix to the gap between cladding and fuel pellets to be released during a cladding breach. The MAR for the postulated fuel handling accident is assumed to equal this gap inventory. This assumption is conservative relative to the RBOF inventory consisting of mostly MTR type fuels. This fuel is essentially a solid matrix (U-Al alloy) fuel with aluminum cladding, which would reduce the amount of fission products available for release during a cladding breach event. Table 3.4-8 shows (1) the fission products that would be released during the accident, (2) the total estimated

radionuclide activity of a single RFA for the nuclides of concern, and (3) the total MAR if the basin was stored with a maximum of 2,800 RFAs.

Damage Ratio

The DR for this scenario is the fraction of material that is released from the fuel matrix into the water. Using DOE-HDBK-3010-94 (Ref. 75), the DR for noble gases and halogens is $5E-01$.

Airborne Release Fraction

The ARF is the coefficient used to estimate the amount of a radioactive material suspended in air as an aerosol. This amount of material would thus be available for transport if no deposition or filtration mechanisms are identified. It is assumed that all H-3 located in the fuel gap at the time of a fuel handling accident would be in vapor form. Again, using DOE-HDBK-3010-94 (Ref. 75), the ARF for the noble gases and halogens through the basin water is $5E-01$. However, all H-3 is assumed to be in the oxide form. Therefore, the ARF is conservatively assumed to be 1.0.

Leak Path Factor

The LPF is the fraction of the radionuclides in the aerosol transported through some confinement deposition or filtration mechanism. For conservatism, the LPF for the fuel handling accident is assumed to be 1.0.

Total Source Term

The total source term is the sum of the MAR, DR, ARF, and LPF. The total source term is provided in Table 3.4-9 and was calculated in S-CLC-H-00399 (Ref. 191).

Consequence Analysis

Radiological consequences resulting from a fuel breach are calculated for the MOI and the individual at the OEP using the methods described in Section 3.4.1 and are summarized in Table 3.4-14. Source terms are derived as described above. Results are substantiated in S-CLC-H-00570 (Ref. 198).

Comparison to Guidelines

The risks associated with a fuel breach are shown in Figure 3.4-2 using the methods described in Section 3.4.1. The risks are a result of the consequences and frequencies summarized in

Table 3.4-14. The radiological risks to the RBOF associated with a high-winds induced fuel breach are within established EGs.

Summary of Safety-Class SSCs and TSR Controls

This accident scenario does not challenge the offsite EGs; therefore, no safety-class SSCs are required.

3.4.2.5 Earthquake

The hazards analysis identified several accident sequences that might occur as the result of an earthquake. These sequences, include the following:

- Potential failure of the facility structure, causing a criticality and breach of fuel
- Loss of pool water, causing the distorted fuel racks and fuel to melt resulting in a criticality
- Waste tank breach, causing the radioactive material to collect on the bottom of the pool and to become resuspended by the air flow
- Loss of chemical material containment in the outside dumpsters

For those sequences discussed under other section headings, the results will only be referenced. The four independent sequences are discussed fully in the sections below.

EARTHQUAKE-INDUCED CRITICALITY

The hazards analysis identified Events 43, 63, and 63a as earthquake-induced criticalities with risk rankings of 3, 5 or 7. The source term is the same bounding source term used for the other criticality events (high winds induced Event 64) discussed in Section 3.4.2.4.

Scenario Development

This accident scenario is initiated by an earthquake, which causes the facility structure and/or racks to fail. The rack failure and/or falling debris impacting the fuel storage racks leads to a critical configuration.

Possible Initiating Events

For this scenario, the initiating event is an earthquake.

Accident Event Progression

Assuming that an earthquake causes the facility to fail, the roof structure and possibly the entire facility could fail, and/or the fuel storage racks could fail. If the racks fail, fuel is potentially dislodged from the racks, and safe spacing is defeated. If the roof sections and potentially the overhead crane collapse into the storage pool, this could result in dislodging the fuel assembly racks and the fuel itself. In either case, the safe spacing of the fuel is disrupted resulting in the potential for a criticality event. Given that the failures cause a criticality, the storage pool water would act to reduce the event's source term. The criticality event is assumed to result in a $5E+18$ fission yield (Section 3.4.1.5).

Preventive Features

Measures to prevent criticality at the RBOF include the following:

- Storage racks are configured and spaced to prevent criticality events. **(Credited)**
- NSDSs specify safe spacing distances between fuel bundles. **(Credited)**

Mitigative Features

The following are mitigative features for this scenario:

- Fuels are stored in water-filled pools under a minimum of 4-1/2 feet of water above the tallest fuel assembly for scrubbing of fission products and attenuation of direct dose from criticality. **(Credited)**
- The NIM system alerts operators that a criticality event with a fission product release is in progress.
- Emergency response plans and onsite medical treatment personnel facilitate removal and treatment of injured workers.
- Site audible warning systems alert workers of the need to evacuate the area.
- Workers are trained to respond safely in the event of a criticality. **(Credited)**

Frequency Binning

The frequency determination for an earthquake-induced criticality at the RBOF is based upon seismic hazard curves for SRS, fragility estimates for structural components at risk, and conditional probabilities of resulting criticality. The magnitude of the initiator is based on criteria established for the PC of the facility. This magnitude establishes the "evaluation basis" that the facility is analyzed against. The frequency of a criticality induced by an evaluation basis earthquake is estimated as $2.55E-08/\text{yr}$, which is in the beyond extremely unlikely ($10^{-6}/\text{yr} > p$)

frequency bin. The analysis is documented in S-CLC-H-00340 (Ref. 188), and is summarized as follows.

The earthquake analyzed is based upon criteria for a PC-3 facility. PC-3 criteria establish that the facility withstand a seismic event with an AEP of $5.0\text{E-}04$. While this may or may not be consistent with the original design basis for the facility, it establishes an evaluation basis against which the facility must be analyzed. Therefore, seismic initiators having an $\text{AEP} \geq 5.0\text{E-}04$ are considered in the determination of evaluation basis criticality frequency, and seismic initiators having an $\text{AEP} < 5.0\text{E-}04$ are considered beyond design/evaluation basis.

An event tree for evaluation basis earthquake-induced criticality was developed and is shown in Figure 3.4-5. The Structural Mechanics Section of PECD provided fragility values for three particular components at risk due to an earthquake: the entire building structure, the fuel racks, and the concrete basin. The concrete basin was identified due to water loss concerns, which are addressed in the subsection associated with the breach of the pool basin. Therefore, the event tree given in Figure 3.4-5 considers whether or not the structure fails during an evaluation basis seismic event (top event NO-STRUCT), whether or not the racks fail during an evaluation basis seismic event (top event NO-RACK), and whether or not the failure or failures result in a criticality (top event NO-CRIT).

An EXCEL spreadsheet was constructed to calculate the event tree sequence frequencies. The spreadsheet is given in S-CLC-H-00340 (Ref. 188). The sequences of concern are SEQ-5S, SEQ-6S, SEQ-9S, SEQ-10S, SEQ-13S, and SEQ-14S. The spreadsheet is set up to calculate the frequency of the seismic event and the probability of success or failure of each event tree branch point for seismic events from 0.001g to approximately 0.13g in 0.001g increments. The upper limit of 0.13g is defined as the seismic event with an AEP of $5.0\text{E-}04$. The AEP of the various earthquake levels is based on the seismic hazard curve for SRS given in T-CLC-G-00024 (Ref. 116).

The deterministic evaluation of the RBOF SSCs of concern is based on the interim site spectrum, which is currently defined as the evaluation basis earthquake as detailed in WSRC-TR-94-0528 (Ref. 206). The interim site spectrum defines a PC-3 seismic event as a 0.19g Zero Period Acceleration (ZPA) event. The RBOF deterministic evaluation provides the basis for which fragility values and associated SSC failure frequencies are derived.

The use of the hazard curve provided in T-CLC-G-00024 for the RBOF probabilistic analysis is acceptable until new hazard curves are developed for the SRS design basis spectrum (Ref. 207).

Although the SRS design basis spectrum has been developed in accordance with DOE-STD-1023-95 (Ref. 208) and is documented in WSRC-TR-97-0086 (Ref. 209), this analysis was not available for use in the structural analysis of the RBOF nor have the associated hazard curves been developed.

As discussed in T-CLC-G-00024, an equation was derived for this curve using a third-order polynomial fit of the data. The equation is as follows, where g is the Peak Ground Acceleration (PGA):

$$AEP = 10^{(-2.2658 - 8.97388 * g + 8.72428 * g^2 - 3.4117 * g^3)} \quad (\text{Eq. 3.4-33})$$

This equation is used to calculate the exceedance probability for each incremental earthquake.

The probability of building structural failure, rack failure, and concrete basin failure is based on fragility estimates for these components and the following formula from EPD-SE-0115 (Ref. 197):

$$P = \phi \left(\frac{\ln \left(\frac{g}{MC} \right)}{\sqrt{\beta_R^2 + \beta_U^2}} \right) \quad (\text{Eq. 3.4-34})$$

where:

- P = probability of failure
- ϕ = standard normal cumulative distribution function
- g = peak ground acceleration
- MC = median capacity of the structural component
- β_R = random uncertainty parameter
- β_U = modeling uncertainty parameter

The median capacity and the uncertainty parameters for the building structure and racks are provided below (Ref. 117, 210). Seismic fragilities for the storage racks were selected based on the assumption that seismic upgrade modifications will be made to the bundle storage rack tie supports and to the main storage rack PM-3A hardware as defined in Chapter 2. In addition, the transfer buckets located in the RBOF disassembly basin and the large test tubes located in the inspection basin have been evaluated against the demands associated with a PC-3 evaluation basis seismic event and were found to be qualified (Ref. 226 and 227).

Structural Component	Median Capacity (MC)	Random Uncertainty Parameter (β_R)	Modeling Uncertainty Parameter (β_U)
Steel Superstructure	0.43g	0.10	0.20
Fuel Racks	0.59g	0.34	0.26

The probabilities of criticality from building structure failure and rack failure were developed based on expert opinion. The opinion elicitation and selection process is documented in S-CLC-H-00350 (Ref. 150) and resulted in the following conditional probabilities:

Probability of criticality given building structure failure = 0.1

Probability of criticality given rack failure = 0.9

Probability of criticality given building structure and rack failure = 0.91

These probabilities were developed consistent with rules for developing split fractions for an Accident Progression Event Tree (Ref. 91).

The overall frequencies for sequences in which a criticality occurs (sequences SEQ-5S, SEQ-6S, SEQ-9S, SEQ-10S, SEQ-13S, and SEQ-14S) are obtained by summing the frequencies calculated for each incremental PGA in the EXCEL spreadsheet. These frequencies are as follows:

SEQ-5S:	2.55E-08/yr
SEQ-6S:	6.73E-14/yr
SEQ-9S:	7.76E-14/yr
SEQ-10S:	2.97E-19/yr
SEQ-13S:	1.18E-16/yr
SEQ-14S	<u>4.81E-22/yr</u>
	2.55E-08/yr

The consequences from a evaluation basis seismic-induced criticality are similar to those from a wind-induced criticality since both events assume the same fission yield. However, since the evaluation basis seismic-induced criticality frequency is in the beyond extremely unlikely bin, no further discussion is provided in this section. Section 3.4.3 provides further discussion of the beyond design basis accidents.

SEISMIC-INDUCED FUEL BREACH

The hazards analysis identified the possibility of a fuel breach as the result of a seismic event causing the facility to fail and the debris to fall into the storage basin and breach fuel bundles. The hazards analysis event number is Event 18.

Scenario Development

In this scenario, the breached fuel would release the fission products contained within the fuel matrix, causing an airborne radionuclide release.

Possible Initiating Events

For this scenario, only one initiating event is examined, a seismic event.

Accident Event Progression

Assuming that an evaluation basis seismic event causes the facility to fail, the roof structure and possibly the entire facility could fail causing roof sections and/or the overhead crane to collapse into the storage basin, rupturing fuel elements. The fission products contained within the fuel matrix would then be released to the environment.

Preventive Features

Preventive features include the following:

- Storage racks will distribute the weight of the debris.
- Fuel storage containers are structurally sound.
- Basin water will reduce the momentum of the falling debris.

Mitigative Features

Mitigative features include the following:

- The required aging of fuel is met prior to receipt at the RBOF. (Credited)
- Basin water will scrub out some of the released material. (Credited)

Frequency Binning

The frequency at which the fuel could breach due to falling debris is assumed to occur at the frequency calculated for the evaluation basis earthquake-induced building and rack failure $2.83\text{E-}08/\text{yr}$. The consequences from a seismic-induced fuel breach are similar to a wind-induced fuel breach since both events assume that the same number of RFAs breach. However, since the evaluation basis seismic-induced fuel breach frequency is in the beyond extremely unlikely bin, no further discussion is provided in this section. Section 3.4.3 provides further discussion.

BREACH OF POOL BASIN

The hazards analysis identified the possibility that the storage basin could be breached in the event of an earthquake, resulting in the basin water draining out. The hazards analysis event numbers are Event 43 and 63a.

Scenario Development

There are two potential failure mechanisms for the concrete basin during a seismic event: (1) basin failure due to ground acceleration, and (2) basin failure due to differential or liquefaction settlement of the soil following a seismic event.. In this scenario, the basin is breached, resulting in the water draining out.

Possible Initiating Events

For this scenario, only one initiating event is examined, an earthquake.

Accident Event Progression

In this event, an earthquake causes a crack in the basin wall, permitting contaminated basin water to leak to the subsoil. This accident is shown to be beyond extremely unlikely for an evaluation basis earthquake in the Frequency Binning subsection.

Preventive Features

A preventive feature for this scenario is that the pool walls and floor are constructed of 2- to 6-foot-thick concrete.

Mitigative Features

A mitigative feature for this scenario is that sensors provide indication if the water level is dropping in the pool.

Frequency Binning

The frequency determination for an evaluation basis earthquake-induced basin draindown at the RBOF is based upon the seismic hazard curves for SRS and fragility estimates for the basin. The acceleration of the initiator is based on criteria established for the PC of the facility. This establishes the "evaluation basis" that the facility is analyzed against. The analysis is documented in S-CLC-H-00340 (Ref. 188) and is summarized as follows.

The earthquake analyzed is based upon criteria for a PC-3 facility. PC-3 criteria establish that the facility withstand a seismic event with an AEP of $5.0E-04$. While this may or may not be consistent with the original design basis for the facility, it establishes an evaluation basis against which the facility must be analyzed. Therefore, seismic initiators having an AEP $\geq 5.0E-04$ are considered in the determination of draindown frequency, whereas seismic initiators having an AEP $< 5.0E-04$ are considered beyond design/evaluation basis.

An event tree for a basin draindown induced by an evaluation basis seismic event was developed. The Structural Mechanics Section of PECD provided fragility values for the concrete basin.

To assess the frequency of a evaluation basis seismic-induced draindown and subsequent fuel heating identified in the hazards analysis, the event tree considers whether or not the basin fails, causing the water to leak to the subsoil during an evaluation basis seismic event (top event NO-DRAIN). There are two potential failure mechanisms for the concrete basin during a seismic event: (1) basin failure due to ground acceleration, and (2) basin failure due to differential or liquefaction settlement of the soil following a seismic event. The first failure mechanism is the only consideration during an evaluation basis earthquake because the settlement mechanism is shown to occur well into the beyond evaluation basis regime in C-CLC-H-00818 and C-CLC-H-00815 (Ref. 211, 212).

C-CLC-H-00818 shows that for a 1/16-inch crack in the basin due to soil settlement, water level losses are only expected to be 9 to 18 inches over a 10-day period. This indicates that it would take a significant amount of time to uncover fuel, since there is normally a minimum of 4.5 feet of water over the fuel. Based on discussions with the Structural Mechanics Section, the crack size is approximately one-half of the soil settlement based on the basin being roughly half as deep as it is long. Therefore, a 1/16-inch crack in the basin is caused by a soil settlement (δ) of approximately 1/8 inch. From C-CLC-H-00815, and assuming an overall length of the basins (L) of approximately 100 feet, an angular distortion (δ/L) of 0.125 inches/100 feet, or $1.0E-04$, has an AEP of roughly $3E-05$. Therefore, based on the water loss analysis from a 1/16-inch crack and the hazard curve for angular distortion, it can be concluded that fuel would not be uncovered due to soil settlement-induced basin cracking at a frequency larger than $3.0E-05/\text{yr}$. In comparison to the AEP of the initiating event for an evaluation basis earthquake ($5.0E-04$), this is beyond the evaluation basis.

An EXCEL spreadsheet was constructed to calculate the event tree sequence frequencies. The spreadsheet is given in S-CLC-H-00340 (Ref. 188). The sequence of concern is SEQ-2DD. The spreadsheet is set up to calculate the frequency of the seismic event and the probability of success or failure of the basin for seismic events from 0.001g to approximately 0.13g in 0.001g increments. The upper limit of 0.13g is defined as the seismic event with an AEP of $5.0E-04$. The AEP of the various earthquake levels is based on the seismic hazard curve for SRS given in T-CLC-G-00024 (Ref. 116).

The deterministic evaluation of the RBOF SSCs of concern is based on the interim site spectrum, which is currently defined as the evaluation basis earthquake as detailed in WSRC-TR-94-0528

(Ref. 206). The interim site spectrum defines a PC-3 seismic event as a 0.19g ZPA event. The RBOF deterministic evaluation provides the basis for which fragility values and associated SSC failure frequencies are derived.

The use of the hazard curve provided in T-CLC-G-00024 for the RBOF probabilistic analysis is acceptable until new hazard curves are developed for the SRS design basis spectrum (Ref. 207).

Although the SRS design basis spectrum has been developed in accordance with DOE-STD-102-95 (Ref. 208) and is documented in WSRC-TR-97-0086 (Ref. 209), this analysis was not available for use in the structural analysis of the RBOF nor have the associated hazard curves been developed.

As discussed in T-CLC-G-00024, an equation was derived for this curve using a third-order polynomial fit of the data. The equation is presented as Equation 3.4-32. This equation is used to calculate the exceedance probability for each incremental earthquake.

The probability of concrete basin failure is based on fragility estimates for the component and upon Equation 3.4-33 (Ref. 197).

The median capacity and the uncertainty parameters for the concrete basin are provided below (Ref. 117) and do not include the effects of soil liquefaction.

Structural Component	Median Capacity (MC)	Random Uncertainty Parameter (β_R)	Modeling Uncertainty Parameter (β_U)
Concrete Basin	2.11g	0.38	0.50

The overall frequency for the sequence in which a basin draindown occurs due to an evaluation basis seismic event is obtained by summing the frequencies calculated for each incremental earthquake in the EXCEL spreadsheet. This frequency is 7.70E-10/yr, which is in the beyond extremely unlikely ($p < 10^{-6}$ /yr) frequency bin. Since this event is beyond extremely unlikely, no further discussion is warranted in Section 3.4.2. Section 3.4.3 provides additional analysis of this beyond evaluation basis event.

BREACH OF WASTE TANK EP38

The hazards analysis identified the possibility that waste tank EP38 could be breached in the event of an earthquake, resulting in the liquid waste draining out. The hazards analysis event number is Event 401.

Scenario Development

In this scenario, waste tank EP38 is breached, resulting in the liquid waste draining out. Left behind would be liquid wasteborne radionuclides.

Possible Initiating Events

For this scenario, only one initiating event is examined, an earthquake.

Accident Event Progression

In the event of an earthquake, the tank is assumed to crack sufficiently to allow for the leakage of liquid waste to the environment. Isotopes entrained within the liquid waste are assumed to dry out and cake on the walls of the tank rather than be drained out with the liquid waste. This analysis assumes 100% caking of the isotopes to the tank walls. Air flow across the dry material leads to resuspension and a conservative estimate of the dose.

Preventive Features

There are no preventive features that reduce the likelihood of this scenario occurring.

Mitigative Features

The following are mitigative features for this scenario:

- Remotely located valve controls are available for shutting off waste flow to the tank in the event of an earthquake.
- Secondary confinement of the waste cell is provided.

Frequency Binning

The frequency determination for an earthquake-induced tank breach is based upon the seismic hazard curves for SRS. The frequency of the event is estimated to be in the unlikely ($10^{-2}/\text{yr} > p > 10^{-4}/\text{yr}$) frequency bin. The analysis is documented in S-CLC-H-00340 (Ref. 188) and is summarized as follows.

T-CLC-G-00024 (Ref. 116) developed a seismic hazard curve for SRS for use in probabilistic risk assessments. The lowest PGA considered is 0.02g. This earthquake has a mean composite AEP of $3.6\text{E-}03$. In the absence of structural analysis for the tank and related equipment involved in this scenario, it is assumed that they would at least withstand an earthquake with a PGA of 0.02g.

For comparison, N-CLC-M-00004 (Ref. 214) reports that a 0.04g PGA corresponds to an earthquake with intensity VI on the Modified Mercalli Intensity Scale. An intensity VI earthquake is described by the following characteristics:

- Persons walk unsteadily.
- Windows, dishes, and glassware break.
- Books and other objects fall off shelves.
- Pictures fall off walls.
- Furniture moves or overturns.
- Weak plaster and poor masonry construction crack.

Based on this information, it is reasonable to assume that the tank and related equipment, although not structurally analyzed, would withstand an earthquake of acceleration 0.02g. Since the initiating event frequency is unlikely ($10^{-2}/\text{yr} > p > 10^{-4}/\text{yr}$), the spill scenario initiated by an earthquake is also at least unlikely.

Source Term Analysis

For this scenario, it is assumed that all of the radioactive material entrained in the liquid waste at the time of tank breach is settled out, either by caking on the walls or the floor. Once dry, air flow across the dry material causes the material to become airborne.

The source term for this scenario was calculated in S-CLC-H-00399 (Ref. 191) and is represented by the equation:

$$ST = MAR * DR * ARF \quad (\text{Eq. 3.4-35})$$

where:

- MAR = amount of isotopes in the liquid waste originally
- DR = amount of material that becomes caked to the walls and floor
- ARF = percentage of material that becomes airborne

Material at Risk

The MAR is the same MAR used in the resin explosion scenario of Section 3.4.2.7 and is shown in Table 3.4-10.

Damage Ratio

The DR is the fraction of isotopes that would cake out of the liquid waste and adhere to the tank walls. Although it would be expected that a significant amount of material would not cake out, for this analysis the DR is conservatively assumed to be 1.0, indicating that all material would cake to the tank.

Airborne Release Fraction

The ARF is provided in DOE-HDBK-3010-94 (Ref. 75). The ARF value is $4\text{E-}06/\text{hr}$. Assuming that it would take 7 days to immobilize the caked powder, the total ARF for the 7 day period would be $6.72\text{-}04$.

Total Source Term

The total source term, by isotope, is provided in Table 3.4-10.

Consequence Analysis

The evaluation basis earthquake for the RBOF produces primary and secondary events. The primary events (criticality, breach of pool basin, and fuel breach) have been shown to be beyond evaluation basis events. The waste tank EP38 breach consequences have been calculated and are reported in Table 3.4-14 and FES-CLC-H-00428 (Ref. 199).

Radiological consequences resulting from an earthquake in the RBOF are calculated for the MOI and the individual at the OEP using the methods described in Section 3.4.1. (The result reported in Table 3.4-14 is represented by only the waste tank breach accident scenario, which will provide bounding radiological consequences for the earthquake event.)

Comparison to Guidelines

The risks associated with an earthquake in the RBOF are shown in Figure 3.4-2 using the methods described in Section 3.4.1. The risks are a result of the consequences and frequencies summarized in Table 3.4-14. The frequency reported for the earthquake is the mean composite AEP of $3.6\text{E-}03$. The radiological risks to the RBOF associated with an earthquake are within established EGs.

Summary of Safety-Class SSCs and TSR Controls

This accident scenario does not challenge the offsite EGs; therefore, no safety-class SSCs are required.

LOSS OF CHEMICAL DUMPSTER CONTAINMENT

The hazards analysis identifies Event 347 as having the potential to breach the chemical storage dumpsters in the event of an earthquake.

Scenario Development

Three chemical dumpsters are located outside of the RBOF building on chemical pads: a 1,000-gallon 50% nitric acid dumpster and 1,000-gallon 50% sodium hydroxide dumpster, both located on chemical pad No. 1, and a 1,600-gallon 30% sodium nitrite hold tank located on chemical pad No. 2. A 1,000-gallon sodium nitrite dumpster is also located on chemical pad No. 2. The dumpsters are replenished with chemicals by disconnecting the piping and transferring the dumpsters to F Area for filling. The dumpsters are then returned to the RBOF and re-connected. In the event of an earthquake, these chemical dumpsters and hold tank have the potential to lose their containment.

Possible Initiating Events

For this scenario, there is only one initiating event, an earthquake.

Accident Event Progression

In the event that an earthquake were to breach a chemical storage dumpster or tank, the contents of the tank would drain to waste tank EP38, assuming that it was still intact and had available volume. The source term calculations presented in this section do not take credit for the waste tanks. In addition, a potential failure mechanism for the dumpster is the collapse of the rain shelter from an earthquake.

Preventive Features

A preventive feature for this scenario is that each dumpster is isolated with a supply valve, which in turn is connected to a flexible hose.

Mitigative Features

The following are mitigative features for this scenario:

- The chemical dumpsters are located in bermed areas that slope to the waste tank EP38 drain, which reduces the amount of chemicals that will evaporate.
- The chemical pad is covered by a rain shelter. The roof does protect part of the pad from direct sunlight, thus reducing the rate of evaporation of spilled chemicals.

- The chemical pads are located in bermed areas that either drain directly to or may be pumped to waste tank EP38. Structural analysis of the chemical pad 2 berm has determined that the berm can withstand the impact from a collapsed rain shelter. The forces placed on the berm are well within the berm's capacity without cracking (Ref. 73, 187) (Credited).

Frequency Binning

For the initiating event of rain shelter collapse, the rain shelter has been structurally evaluated against the seismic loads associated with a PC-2 seismic event or 0.15g ZPA with an AEP of $1\text{E-}03/\text{yr}$. On the basis of the structural analysis documented in T-CLC-H-00244 (Ref. 117), the frequency of the rain shelter collapsing due to seismic events less than or equal to 0.15g ZPA is $2.80\text{E-}14/\text{yr}$ (Ref. 195). Seismic event initiators having an AEP of less than $1\text{E-}03/\text{yr}$ are considered beyond the evaluation basis and are not analyzed.

For the potential scenario where the earthquake does not damage the rain shelter but alone could potentially damage the dumpster, a structural analysis of the dumpster and hold tank and supporting concrete slab has been performed. This analysis qualified the dumpster and slab to all loads associated with PC-2 NPH events including earthquake and wind loads (Ref. 73, 217). Seismic event initiators having an AEP of less than $1\text{E-}03/\text{yr}$ are considered beyond the evaluation basis and are not analyzed.

In lieu of crediting the structural resistance of the rainshelter to seismic loads, the consequences resulting from an assumed rainshelter failure were evaluated. This scenario assumes the rainshelter would withstand an earthquake of acceleration of 0.02g similar to the logic used to assess the structural resistance of the waste tank EP38.

Source Term Analysis

The source terms for the three possible chemical spills (50% nitric acid, 50% sodium hydroxide, and 30% sodium nitrite) are the same as they are reported in Section 3.4.2.1. The release under the freezing conditions is expected to yield lower source terms due to the conditions outlined under the heading "Mitigative Features" above. Therefore, the scenario used to calculate the source terms in Section 3.4.2.1 is considered bounding for this low temperature scenario.

Section 3.4.2.1 indicates that sodium hydroxide will be disbursed at a rate of $3.5\text{E-}06$ kg/sec, nitric acid will be disbursed at a rate of 0.0673 kg/sec, and sodium nitrite will be disbursed at a rate of $4\text{E-}06$ kg/sec.

Consequence Analysis

The consequences are the same as those presented in Section 3.4.2.1. For the sodium nitrite and sodium hydroxide releases, the consequences of the release are insignificant and there is no need

for further analysis at greater distances. The consequences for the receptors located at the OEP and at the site boundary are acceptable. (Refer to the hazardous chemical worker safety section of Section 3.3.2.3 for the onsite receptor consequence methodology and analysis).

Comparison to Guidelines

Based on the frequency for this event, the comparison to EGs is based on the analysis in Section 3.4.2.1.

Summary of Safety-Class SSCs and TSR Controls

This accident scenario does not challenge the offsite EGs; therefore, no safety-class SSCs are required. (Refer to the hazardous chemical worker safety section of Section 3.3.2.3 for the onsite receptor consequence methodology and analysis).

3.4.2.6 Fire

The bounding fire accident identified in the hazards analysis is a resin fire involving the portable deionizers in the RRF(Event 310. The hazards analysis categorized the risk of this event as having moderate consequences and an anticipated likelihood of occurrence, or a risk rank of 4.

SCENARIO DEVELOPMENT

Spent portable deionizers from the 100 Area are regenerated in the regeneration cell of the RBOF and RRF. These deionizers contain contaminated polystyrene resins (combustible). After a breach, the contaminated resin comes in contact with an ignition source with sufficient energy to dry and ignite the resin.

POSSIBLE INITIATING EVENTS

The following are possible initiating events for this scenario:

- Material failure of the deionizer tanks, associated pipes, valves, or hoses coupled with an ignition source.
- High pressure resulting in a breach of containment coupled with an ignition source.
- Human errors resulting in a loss of containment coupled with an ignition source.

ACCIDENT EVENT PROGRESSION

The resin leaks from the portable deionizer, dries and is then exposed to an ignition source with sufficient energy to ignite the resin.

PREVENTIVE FEATURES

The following are preventive features (none of which are credited in the analysis):

- The resin will have to dry to the point the ignition source has enough energy to ignite the resin.
- There are no ignition sources (with needed energy) located at the RRF.

MITIGATIVE FEATURES

The following are mitigative features (none of which are credited in the analysis):

- Much of the resin is not likely to burn because of obstructions around the resin or from the vessel.
- Deposition of the airborne particles will reduce the source term.

FREQUENCY BINNING

A fault tree was developed to quantify the frequency of a portable deionizer resin fire due to a loss of resin containment (i.e., leak, equipment malfunction, operator error) coupled with an ignition probability). The fault tree analysis is documented in S-CLC-H-00281 (Ref. 179). The frequency of the event, as estimated by the fault tree, is $8.15\text{E-}04/\text{yr}$, which is in the unlikely ($10^{-2}/\text{yr} > p > 10^{-4}/\text{yr}$) frequency bin.

The fault tree was developed and quantified using the CAFTA computer code (Ref. 180). Input data for each basic event consisting of component failure rates and human error probabilities, are obtained from References 60, 108, 120, 181, and 182. The logic of the fault trees is arranged into three major categories: material failure, high pressure, and human error.

The following assumptions were used as input that are specific to the fault tree evaluation performed for a portable deionizer resin fire:

- The facility is operated 24 hours a day, 365 days a year. The portable deionizer is always in operation, except when removed from the system for refilling.
- Portable deionizer EP216 is welded to a flatbed truck. No credit is taken for the ability of any diked area to contain and collect a resin spill.

- Independent verification is not performed on proceduralized activities involving connection and disconnection of the portable deionizer at the hose rack located to the west of the RRF.
- A cumulative total of 30 feet of piping is connected between the portable deionizer and the valves installed in the valve room.
- A total of 10 feet of flexible hose is connected between the portable deionizer and the first valve installed on the hose rack.
- Failure due to structural fatigue is not considered because the portable deionizer and associated piping and valves are not subjected to cyclic high pressure transients or cyclic structural loading in excess of the fatigue limit (i.e., 2 to 10 million cycles) (Ref. 183) for most steels. Although corrosion due to caustic and acidic chemicals can substantially reduce fatigue strength, the low frequency of valve manipulations, involving connection and disconnection of hoses and vessel movements associated with refilling activities, do not warrant analysis for corrosion fatigue.
- Failure due to thermal stress is not considered because the portable deionizer and associated piping and valves are not normally subjected to cyclic high temperature transients.
- Typically, 20 to 30 resin regenerations using the portable deionizer occur every year. A maximum of 50 regeneration activities using the portable deionizer can occur each year.
- Actual operating experience shows that blockage of resin line #225 occurs approximately 1 in every 25 regeneration operations.
- A pressure surge in the plant air system occurs with the same frequency as a pressure surge from the deionized water pump. This assumption is conservative because the frequency of failure for most types of pumps is an order-of-magnitude higher than for most types of air compressors (Ref. 181). This pressure surge pressurizes the deionizer flex hoses above design capacity.

The two dominant scenarios (i.e., the cutsets contributing the most to a potential resin fire) are represented by the cases when an operator fails to properly connect the resin-out and water-out flex hoses due to human error. The operator fails to stop the spill and the resin ignites. A conditional probability of 0.1 is assumed for the resin igniting based on ignition source studies performed for other facilities.

SOURCE TERM ANALYSIS

A fire that burns the polystyrene resin from a portable deionizer will release a portion of the radionuclide inventory that is attached to the resin.

Material at Risk

Deionizers and the resin they contain are used in the 100 Area to reduce the contamination storage basins. Table 3.4-13 represents the bounding activity of a portable deionizer sent to the RBOF and RRF for regeneration.

Source Reduction Factors

The DR is the fraction of the MAR that is impacted by the accident-generated conditions. For this scenario, the DR translates into the amount of material that would be involved in the fire. The scenario assumes that all of the material is involved in the fire, and the DR is conservatively set to 1.0. Section 5.2.1.4 of DOE-HDBK-3010-94 (Ref. 75) recommends a bounding ARF for the resin fire to be $1\text{E-}02$ for all isotopes (with the exception of H-3, which has an ARF of 1.00) and the RF to be 1.0 based on a contaminated polystyrene fire. The LPF is the percentage of material that would be entrained as the plume. For this scenario, it is assumed that the LPF is 1.0.

Source Term

To find the source term, the MAR for each radionuclide in Table 3.4-13 is multiplied by the DR ($=1.0$), the ARF ($=1\text{E-}02$), the RF ($=1.0$), and the LPF ($=1.0$). The results are calculated in S-CLC-H-00399 (Ref. 191) and are presented in Table 3.4-13.

CONSEQUENCE ANALYSIS

Radiological consequences resulting from a resin fire are calculated for the MOI and the individual at the OEP using the methods described in Section 3.4.1 and are summarized in Table 3.4-14. Source terms are derived as described above. Results are substantiated in S-CLC-H-00570 (Ref. 198).

COMPARISON TO GUIDELINES

The risks associated with a resin fire are shown in Figure 3.4-2 using the methods described in Section 3.4.1. The risks are a result of the consequences and frequencies summarized in Table 3.4-14. The radiological risks to the RBOF associated with a resin fire are within established EGs.

SUMMARY OF SAFETY-CLASS SSCs AND TSR CONTROLS

This accident scenario does not challenge the offsite EGs; therefore, no safety-class SSCs are required.

3.4.2.7 Explosion

The bounding explosion accident identified in the hazards analysis is an accidental mixing of concentrated nitric acid and resin in the regeneration cell. The hazards analysis categorized the risk of this event as having low consequences and an anticipated likelihood of occurrence, or a risk rank of 7.

SCENARIO DEVELOPMENT

Spent resin from both the RBOF and the 100 Area is regenerated in the regeneration cell of the RBOF and RRF. The spent resin is first depleted in tank EP57 using dilute sodium hydroxide and then separated using deionized water and air. The anion resin is then transferred to tank EP59 for regeneration with sodium hydroxide while the cation resin is regenerated with nitric acid in tank EP57.

The regeneration cell is constructed of 36-inch-thick concrete walls, 24-inch concrete floor, and a ceiling made of mostly unprotected metal decking with removable wooden covers over the tanks. Sodium hydroxide and nitric acid are both pumped to tank EP57.

The accident scenario of concern is an explosion due to the addition of undiluted nitric acid to the resin bed. Undiluted acid promotes rapid resin degradation.

POSSIBLE INITIATING EVENTS

The following are possible initiating events for this scenario:

- Operator error allows concentrated nitric acid to enter tank EP57 for the separation process, coupled with a failure of the deionized water supply or human error resulting in failure of the deionized water to be supplied to tank EP57.
- The sodium hydroxide and nitric acid dumpsters are erroneously switched.
- The sodium hydroxide and nitric acid dumpsters are incorrectly filled in F Area.

ACCIDENT EVENT PROGRESSION

Once the initiating event occurs, the concentrated nitric acid mixes with the resin to cause either an overpressure situation or an explosion. The reaction breaches the tank and the roof material since it is the weakest barrier of the cell. The radionuclides in the resin are then dispersed into the atmosphere.

PREVENTIVE FEATURES

The following are preventive features for this scenario:

- Administrative controls specify which valves to open and close to ensure that diluted nitric acid is used for regeneration.
- Operators are experienced in filling the nitric acid and sodium hydroxide dumpsters, and different size fittings are used on the fill lines, thereby reducing the likelihood of filling a dumpster with the wrong material.

MITIGATIVE FEATURES

The following are mitigative features for this scenario:

- The regeneration cell is constructed of thick concrete, which will reduce the total amount of radionuclides released since some of the material will adhere to the walls and floor.
- The building exhaust filter may entrain some of the radionuclides prior to roof failure.
- Limitations on the ion loading of the resin will limit the maximum total activity possible for the explosion. (Credited)

FREQUENCY BINNING

A fault tree was developed to quantify the frequency of a resin explosion due to the initiating events listed above. The fault tree analysis is documented in S-CLC-H-00281 (Ref. 179). The frequency of the event, as estimated by the fault tree, is $2.50\text{E-}03/\text{yr}$, which is in the unlikely ($10^{-2}/\text{yr} > p > 10^{-4}/\text{yr}$) frequency bin.

The fault tree was developed and quantified using the CAFTA computer code (Ref. 180). Input data for each basic event, consisting of component failure rates and human error probabilities, are obtained from References 60, 107, 108, 181, and 182. The logic of the fault trees is arranged into two major categories: (1) loss of deionized water due to equipment malfunction, and (2) loss of deionized water due to human error. The scenario related to inadvertently switching the sodium hydroxide dumpster with the nitric acid dumpster is not addressed in the fault tree because it is not considered a significant contributor to the overall frequency. The contents of the sodium hydroxide dumpster are heated with steam; thus, the reconnection configuration differs from that of the acid dumpster, which is not heated.

Another scenario considered in the analysis involves the addition of excess nitric acid during resin regeneration while maintaining normal deionized water flow. By procedure, nitric acid is set to deliver between 1.1 and 1.6 gal/min of acid for 59 minutes (or 64.9 to 94.4 gallons for each of two pumps). The maximum capability of each pump is 1.8 gal/min for 59 minutes (or

106.2 gallons). The excess addition of 11.8 gallons of acid per pump is not considered to be of concern provided that normal deionized water supply is maintained as a dilution agent. Thus, the critical parameter leading to a potential resin explosion is the loss of deionized water, regardless of normal or abnormal nitric acid feed with existing acid pump capabilities.

The following assumptions were used as input and are specific to the fault tree evaluation performed for a resin explosion:

- The facility is operated 24 hours a day, 365 days a year.
- Regeneration of the resin in the RBOF facility occurs a maximum of three times each year (typically, twice each year).
- Regeneration of the resin shipped via the portable deionizer occurs a maximum of 50 times each year (typically, 20 to 30 times each year).
- The acid dumpster is filled with the correct chemical when shipped for refilling. On the basis of S-CLC-H-00479 (Ref. 218), the probability of filling the tanks incorrectly is $5.0\text{E-}07$, which is in the beyond extremely unlikely ($p < 10^{-6}/\text{yr}$) frequency bin.
- It is considered highly unlikely that the acid and caustic dumpsters can be switched on the chemical pad at the time of delivery. The contents of the caustic dumpster are heated with steam; thus, the reconnection configuration differs from that of the acid dumpster, which is not heated.

The dominant scenario (i.e., the cutset contributing the most to a potential resin explosion) is represented by the case where an operator fails to open deionized water valve DI-3 and the independent checker fails to correct the error.

SOURCE TERM ANALYSIS

The source term for the resin explosion is the product of the activity on the resin at the time of the explosion (MAR), the percentage of material that leaks from the tank (DR), the percentage of radionuclides that are lifted airborne as a result of the fire (ARF), and the fraction of material that escapes from the building confinement, either up through the ventilation exhaust stack or by some other means (LPF). This equation is presented as Equation 3.4-25.

Material at Risk

To determine the source term for the resin fire, the MAR of the RBOF resin is first calculated. The equation for the MAR is the product of the average activity in the water, the rate at which the water flows through the deionizer, and the duration prior to resin bed regeneration.

Once the total activity of the resin is determined, the isotope fractions are calculated using the values in Table 3.4-11. The isotopic fraction in the resin bed is based on the isotopic fraction identified in laboratory samples of resin waste and the resin bed outlet (Ref. 82):

The total activity of the resin after 6 months of operation is as follows:

$$\begin{aligned}\text{Activity} &= (1\text{E-}02 \text{ }\mu\text{Ci/mL}) * (1\text{E-}06 \text{ Ci/}\mu\text{Ci})(1,000 \text{ mL/L}) * (3.785 \text{ L/gal}) * \\ &\quad (150 \text{ gal/min}) * (60 \text{ min/hr}) * (24 \text{ hr/day}) * (5 \text{ days/wk}) * (26 \text{ weeks}) \\ &= 1,060 \text{ Ci}\end{aligned}$$

This number is multiplied by the isotopic fractions provided in Table 3.4-11 to derive the MAR provided in Table 3.4-12.

The total activity for the RBOF resin exceeds the 100-Area resin, which historically has had an activity of 0.729 Ci at the time of shipment to the RBOF and RRF for regeneration (Ref. 59). Due to longer run times in the 100 Area as a result of the deionizer modifications, the loading could possibly increase but would not approach the value of 1,060 Ci assumed here.

Damage Ratio

The DR is the fraction of the MAR that is impacted by the accident-generated conditions. For this scenario, the DR translates into the amount of material that would be released from the tank in the event of a breach or leak. This scenario assumes a total release of material. The DR is conservatively estimated to be 1.0.

Airborne Release Fraction

The ARF is provided in Section 5.0 of DOE-HDBK-3010-94 (Ref. 75). For an overpressure event or blast event, the ARF is 1E-03.

Leak Path Factor

The LPF is the percentage of material that would be entrained as the plume exited the facility. In this event, the blast would be expected to damage the roof and bypass the ventilation system and the exhaust filter. Therefore, for this scenario, it is conservatively assumed that the LPF is 1.0.

Total Source Term

To find the source term, the MAR for each radionuclide is multiplied by the DR (=1.0), the ARF (=1E-03), and the LPF (=1.0). The results are calculated in S-CLC-H-00293 (Ref. 219) and are presented in Table 3.4-12.

The total source term for a resin explosion is 1.06 Ci.

CONSEQUENCE ANALYSIS

Radiological consequences resulting from a resin explosion are calculated for the MOI and the individual at the OEP using the methods described in Section 3.4.1 and are summarized in Table 3.4-14. Source terms are derived as described above. Results are substantiated in S-CLC-H-00318 (Ref. 215).

COMPARISON TO GUIDELINES

The risks associated with a resin explosion are shown in Figure 3.4-2 using the methods described in Section 3.4.1. The risks are a result of the consequences and frequencies summarized in Table 3.4-14. The radiological risks to the RBOF associated with a resin explosion are within established EGs.

SUMMARY OF SAFETY-CLASS SSCs AND TSR CONTROLS

This accident scenario does not challenge the offsite EGs; therefore, no safety-class SSCs are required.

3.4.2.8 Uncontrolled Chemical Reaction

SCENARIO DEVELOPMENT

The RBOF is a facility used for the safe handling and storage of fuels of various shapes, sizes, and content. In addition to receiving sound fuel assemblies, the RBOF may be asked to receive unusual materials, such as failed fuel assemblies shipped within special vented capsules, sodium and potassium bonded assemblies shipped within sealed capsules, or fuel assemblies that have failed enroute to the storage facility. This scenario involves the sodium and potassium bonded fuels stored within sealed capsules. An uncontrolled chemical reaction is postulated to occur between fuel that is improperly stripped of sodium or potassium and the basin water due to a leaking container.

ACCIDENT EVENT PROGRESSION

Once the initiator occurs (leaking container), the basin water reacts with sodium or potassium that was improperly stripped of the fuel prior to being shipped to the RBOF. The volatile fission products escape the breached container into the basin. These fission gases are not trapped by the basin water and are released to the atmosphere. When the sodium and potassium react with the water, they form sodium hydroxide or potassium hydroxide, respectively. Most of the potassium hydroxide or sodium hydroxide and other nonvolatiles are retained in the basin water.

PREVENTIVE FEATURES

The following are preventive features for this scenario:

- The sodium bonded fuels are stored in sealed containers.
- The offsite shipper is required to completely strip the sodium (or other materials that react violently with water) from the fuel. **(Credited)**

MITIGATIVE FEATURES

A mitigative feature for this scenario is that the basin water will reduce the release of nonvolatile radioactive components that escape the fuel.

FREQUENCY BINNING

A fault tree was developed to quantify the frequency of an uncontrolled chemical reaction between sodium and potassium bonded fuels and basin water. The fault tree analysis is documented in S-CLC-H-00281 (Ref. 179). The frequency of the event, as estimated by the fault tree, is $2.16\text{E-}03/\text{yr}$, which is in the unlikely ($10^{-2}/\text{yr} > p > 10^{-4}/\text{yr}$) frequency bin.

The fault tree was developed and quantified using the CAFTA computer code (Ref. 180). Input data for each basic event, consisting of component failure rates and human error probabilities, are obtained from References 107, 108, 181, and 182. The fault tree calculates the frequency per year for an uncontrolled chemical reaction due to credible causes, such as corrosion and dropping of a fuel container due to equipment malfunction and human error.

The following assumptions were used as input and are specific to the fault tree evaluation performed for an uncontrolled chemical reaction:

- A total of 37 SRE and 59 Experimental Breeder Reactor (EBR-II) cans are currently stored in the RBOF. The SRE and EBR-II fuels are sodium and potassium bonded fuels that have been chemically stripped of sodium and potassium and sealed in single-wall aluminum cans. Disposition of this material has not yet been determined; however, the fault tree analysis assumes that all 96 cans will be shipped to another facility during a year's period.
- The shipper of the offsite fuel is required to fully strip the fuel of sodium and potassium bonding. The fault tree analysis accounts for errors that could have been made during this stripping process.
- Water is always present in the event of a fuel container failure, and water ingress into the failed containers causes a violent reaction between the sodium or potassium and water if the fuel is improperly stripped.

- Although final disposition of these fuels has not yet been determined, the fuels will all be shipped to another facility during a 1-year period. Further, these shipments would be accomplished with 10 casks.
- A chain hoist is used to lift one container at a time and to transfer each container from a storage basin to another working basin in the RBOF. A chain hoist operates for 30 minutes for each operation of container handling/movement (i.e., a chain hoist bears the fuel container load for 30 minutes during each operation).
- A crane is used in cask handling operations. A crane operates for 2 hours for each operation of cask handling/movement (i.e., a crane bears the cask load for 2 hours during each operation).
- Any potential gas generated inside an offsite fuel container due to fuel decay, radiolysis, or oxidation would not result in the magnitude of fuel damage assumed to be caused by a reaction of sodium or potassium and basin water.
- Independent checking is not performed on proceduralized activities involving fuel container movement.

The dominant scenario (i.e., the cutset contributing the most to an uncontrolled chemical reaction) is represented by the case where a fuel container leaks due to corrosion, which introduces water to the fuel that was improperly stripped of sodium or potassium.

SOURCE TERM ANALYSIS

The radionuclide source term for the breach of sodium fuel is bounded by the analysis conducted in S-CLC-H-00399 (Ref. 191) and presented in Section 3.4.2.4. The source term presented in Section 3.4.2.4 is based on the assumption that all 2,800 fuel storage locations within the pool are holding fuel and that all of the fuel is breached. In addition, the fuel composition is the RFA, which is defined in N-CLC-H-00084 (Ref. 21) as a worst-case fuel based on the highest isotope loading from all of the potential fuels that might be stored at the RBOF.

CONSEQUENCE ANALYSIS

Radiological consequences resulting from an uncontrolled reaction are calculated for the MOI and the individual at the OEP using the methods described in Section 3.4.1 and are summarized in Table 3.4-14. Source terms are derived as described above. Results are substantiated in S-CLC-H-00570 (Ref. 198).

COMPARISON TO GUIDELINES

The risks associated with an uncontrolled reaction are shown in Figure 3.4-2 using the methods described in Section 3.4.1. The risks are a result of the consequences and frequencies

summarized in Table 3.4-14. The radiological risks to the RBOF associated with an uncontrolled reaction are within established EGs.

SUMMARY OF SAFETY-CLASS SSCs AND TSR CONTROLS

This accident scenario does not challenge the offsite EGs; therefore, no safety-class SSCs are required.

3.4.2.9 Aircraft Impact

The frequency determination for aircraft impacts at the RBOF and RRF is based upon estimated airplane/helicopter traffic in the vicinity of the RBOF coupled with estimated accident frequencies/mile traveled. The frequency is estimated as $7.40\text{E-}07/\text{yr}$, which is in the beyond extremely unlikely ($p < 10^{-6}/\text{yr}$) frequency bin. Since this event is beyond extremely unlikely, no further discussion is warranted. The frequency analysis is documented in S-CLC-H-00281 (Ref. 179) and is summarized as follows.

Postulated aircraft impacts for the RBOF include impacts by SRS (Wackenhut) helicopters and offsite (commercial and military) airplanes. The frequency bin determination for aircraft impacts at the RBOF is based, in part, upon a similar determination made for the SWMF (Ref. 121).

The following data are used as input in the evaluation of aircraft impact events:

- The helicopter crash frequency per unit area is $5.3\text{E-}11/\text{yr-ft}^2$ (determined for the SWMF in WSRC-RP-89-715 [Ref. 121]).
- The airplane crash probability per flight per unit area is $4.0\text{E-}09$ per flight per square mile (determined for SRS in WSRC-RP-89-715).
- The number of airplanes entering SRS air space is estimated to be 13,870 flights per year (Ref. 121).
- The approximate area occupied by the RBOF and adjacent structures is $2.6\text{E+}04 \text{ ft}^2$, or $9.2\text{E-}04 \text{ mi}^2$ (conservatively estimated from site maps).

The following assumptions are used as input in the evaluation of aircraft impact events:

- The helicopter crash frequency determined for the SWMF (in occurrences per year per square foot) (Ref. 121) is area-dependent, not facility-dependent. Since the RBOF, RRF, and SWMF are located in the same area of the site, it is reasonable to assume the SWMF value as being applicable to the RBOF.
- The helicopter crash frequency per unit area of $5.3\text{E-}11/\text{yr-ft}^2$ determined for the SWMF does not take credit for the fact that helicopter flight plans purposely avoid overflights to minimize the potential for crashing into the facilities. A factor of 0.5 is assumed in this calculation to account for this fact.

The frequency bin determination for aircraft impacts is based upon the sum of the results for helicopter and airplane crashes.

HELICOPTER CRASH FREQUENCY

WSRC-RP-89-715 (Ref. 121) calculated the helicopter crash frequency for the transuranic pads located in the SWMF to be $5.3\text{E-}11/\text{yr-ft}^2$. This crash frequency is based upon the assumption that the crash probability for the transuranic pads is the same as for H Area due to the proximity and similarity of terrain. This crash frequency is also based upon helicopters that are representative of those used at SRS and upon 770 overflights per year. Since the RBOF and RRF are located inside H Area with similar terrain, it is reasonable to assume the same crash frequency for the RBOF and RRF (i.e., $5.3\text{E-}11/\text{yr-ft}^2$). This frequency is conservative due to the following reasons:

- No credit is taken for the fact that helicopters purposely avoid flying directly over major processing facilities, such as the RBOF and RRF, to minimize the potential for crashing into these facilities.
- Training and practice exercises (which comprise the majority of the flight hours) frequently require landing in an open area away from major facilities.

Thus, a factor of 0.5 is assumed in this calculation to account for the fact that helicopters avoid flying directly over facilities. This factor is conservative because it assumes that 50% of the flights do fly over major facilities. The approximate area of the RBOF is $2.6\text{E+}04 \text{ ft}^2$ (approximated from site maps). The resulting helicopter crash frequency for the RBOF is calculated as

$$(5.3\text{E-}11/\text{yr-ft}^2)(2.6\text{E+}04 \text{ ft}^2)(0.5) = 6.89\text{E-}07/\text{yr}.$$

AIRPLANE CRASH FREQUENCY

S-CLC-E-00048 (Ref. 122) calculated the aircraft crash probability for the SWMF to be $4.0\text{E-}09/\text{flight-mi}^2$. This crash probability is based upon the SWMF being located more than 10 miles from any airport runway. S-CLC-E-00048 also identifies the total number of commercial, military, and general aviation flights over SRS to be 13,870 per year. It is reasonable to apply these same values to the RBOF and RRF in this calculation due to the proximity of the RBOF and RRF to the SWMF.

Since the total target area for the RBOF and RRF is $9.2\text{E-}04 \text{ mi}^2$, the expected frequency of an airplane crash at the RBOF and RRF is calculated as

$$(4.0\text{E-}09/\text{flight-mi}^2) (13,870 \text{ flights/yr})(9.2\text{E-}04 \text{ mi}^2) = 5.10\text{E-}08/\text{yr}.$$

OVERALL AIRCRAFT IMPACT FREQUENCY

The overall aircraft impact frequency is the sum of the helicopter and airplane crash frequencies, or

$$(6.89\text{E-}07/\text{yr}) + (5.10\text{E-}08/\text{yr}) = 7.40\text{E-}07/\text{yr}.$$

3.4.2.10 Selection of Bounding Accident for the Anticipated Frequency Bin

As part of the safety analysis procedure set forth by DOE Order 5480.23 (Ref. 1), a hazards analysis is a requirement for the RBOF and RRF to comprehensively identify and characterize all known hazards and to evaluate them in terms of actual facility operations to assess potential risk. The hazards analysis provides a basis for hazard classification, which in turn determines the level of safety documentation required for the facility and the level of review and approval required for the safety documentation.

Sections 3.1 through 3.3 provided the results of the hazard identification, hazard classification, and hazard evaluation for the RBOF and RRF. Items discussed in these sections included the following:

- Methodology for hazard and accident analysis
- Identification of hazardous energy sources and hazardous material sources present
- Radiological and chemical inventories
- Facility hazard classification
- Risk evaluation of identified accident scenarios based on a qualitative assessment of consequences and frequency
- Summary of design and operational preventive and mitigative features
- Identification of planned design and operational safety improvements
- Summary of defense in depth, including identification of safety-significant SSCs and other items needing TSR coverage
- Summary of significant worker safety features, including identification of safety-significant SSCs and any relevant TSR administrative controls
- Identification of a limited set of unique and representative accidents to be analyzed further

The final step in the hazard evaluation process was to determine the risk rank for each postulated event. This risk rank prioritizes accident scenarios for review or further analysis and was included in the hazard evaluation tables.

Risk rankings are associated with sectors of the risk matrix (Table 3.3-10) in a manner that prioritizes accident scenarios for review or further analysis primarily in terms of the risk that they present, with the consideration that scenarios in the low consequence sectors should represent no more than a marginal concern, regardless of the frequency. Risk rankings are identified and distinguished based upon consequences to the MOI and the onsite worker.

The risk rank for each postulated event was determined by locating the event on a risk matrix based upon the consequence level and frequency level of the event. The risk rank for each event is identified based upon the consequences associated with the MOI and the onsite worker. This limited set of unique and representative accidents was analyzed in detail in Section 3.4.

To provide a tool for evaluating the risk of proposed activities or discoveries against the risk that has been authorized by DOE, the results of the above quantitative accident analysis are plotted as shown on Figure 3.4-2, referred to as a "Farmer Plot." This plot uses the consequences and frequency from each event to provide a bounding risk level for each frequency bin. As a result of this quantitative analysis for the RBOF and RRF, all of the accidents that were categorized as "anticipated" events (estimated annual probability of occurrence of greater than $1\text{E-}02$) and quantitatively evaluated for event frequency were found to have frequencies of less than $1\text{E-}02/\text{yr}$.

To provide a bounding consequence value for the "anticipated" frequency bin for use in the farmer plot, a qualitative review of the events with risk ranks of 1, 4, or 7 was performed. From Table 3.3-12, there are no risk rank 1 events for the RBOF. The risk rank 4 events were all quantitatively analyzed in Section 3.4. Therefore, the candidates for a bounding anticipated event were limited to the risk rank 7 events. Of these, and based on this qualitative review, the bounding accident is considered to be Event 11, which is an airborne release from a breached or failed fuel assembly due to material failure, thermal stress, corrosion, chemical reaction, or mechanical damage.

From Table E-1 of S-CLC-H-00203 (Ref. 25), the frequency of a fuel breach is estimated to be $7.23\text{E-}01/\text{yr}$. The consequences from a breached or failed fuel assembly will be estimated by assuming that 10% of the entire inventory of RFAs (280 RFAs) have failed with 10% of the source term provided in Table 3.4-9 being released. Consistent with previous analysis for the fuel breach event, no credit will be assumed for scrubbing of the released fission products by the basin water with the ARF and LPF set to 1.0. The doses for the fuel breach event are provided in Table 3.4-14. Using the previous assumption that 10% of the RFA inventory is affected, the resulting doses are 10% of the values defined for the fuel breach scenario or 25 mrem for the individual at the OEP (at 640 meters) and 3.3 mrem for the MOI. Table 3.4-14 and Figure 3.4-2 include the values to provide a bounding risk value for the anticipated frequency bin.

3.4.3 BEYOND DESIGN BASIS ACCIDENTS

DOE Order 5480.23 (Ref. 1) requires the evaluation of accidents beyond the design basis to provide a perspective of the residual risk associated with the operation of the facility. The

Beyond DBAs are not analyzed to the same level of detail as DBAs, but rather simply provide insight into the magnitude of consequences of Beyond DBAs. This insight from Beyond DBA analysis has the potential for identifying additional facility features that could prevent or reduce severe Beyond DBA consequences. Only Beyond DBAs of significance (for internal events and NPHs) are identified and assessed for their importance (Ref. 2). For nonreactor nuclear facilities, the sharp increase in consequences from DBA to Beyond DBA is not anticipated to approach that found in commercial reactors where the Beyond DBA precedent was generated (Ref. 1, 2).

For beyond design basis seismic activities, two potential accident scenarios are possible. The first is an extension of the design basis scenario (earthquake-induced criticality), and the second is an earthquake-induced basin structural failure (basin draining) followed by partial fuel melting.

3.4.3.1 Potential Earthquake-Induced Criticality

As the severity of the postulated seismic event increases (frequency decreases), the damage to the building, storage racks, and other components also increases. Thus, the conditional probability of a criticality after an earthquake could increase.

Consequences from a criticality induced by a beyond design basis earthquake would remain the same as for those for the design basis event.

3.4.3.2 Potential Earthquake-Induced Fuel Melting

The RBOF basin has substantial capability to withstand seismic events with magnitudes beyond the design basis / evaluation basis earthquake. However, if the severity of the earthquake increases significantly, the following potential fuel melt scenarios cannot be precluded (Ref. 220):

- At very high earthquake PGAs the integrity of the basin may be jeopardized. At PGAs with a probability of $1E-05/\text{yr}$, significant differential settlement cannot be precluded. The structural consequences of soil liquefaction may be cracks in the basin (allowing the water to drain) and substantial structural and component damage. Water draining could take days depending on crack size.
- Even without water in the basin, fuel melting and subsequently releasing radionuclides is not expected with the original geometric storage arrangement of the basin. However, if sufficient structural and component damage occurs, some of the fuel may be packed into local regions and an insulating debris layer may be formed over part of it. Some local fuel heating and melting could then occur. Considering the localized nature of compaction and the potential insulating debris, as well as the low decay heat and its distribution among the assemblies within the RBOF, only a small fraction of the fuel, if any, is expected to melt.

- The melting process is expected to occur over some time (days) because of the thermal considerations mentioned above and because the molten fuel droplets would flow to cooler regions where the other fuel assembly materials, handling tubes, storage racks, and debris would provide sufficient heat sink to freeze the fuel.
- Most of the noble gases and many of the halogens and volatile solids would be expected to be released during the melting process. Very little, if any, of the other radionuclides would be released.
- After being released from the fuel, most of the nonvolatile radionuclides and considerable amount of the halogens and volatiles would deposit onto the surrounding cooler debris surfaces. If the fuel concentration and insulating debris are sufficiently compact to prevent convective air cooling, they are also sufficient to trap many fission product vapors and aerosols for enough time to allow aerosol settling and deposition on the debris.
- The remaining radionuclides are released to the atmosphere over several days. This release duration increases the dispersion compared to the typical design basis short time dispersion characteristics, which are much more conservative.
- During transport of the radionuclides be the atmosphere, considerable deposition of the non gaseous radionuclides will occur.

The resulting potential doses, evaluated using the above scenario with best-estimate parameters, are within the EGs for extremely unlikely DBAs.

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3.6 TABLES

Table 3.3-1 Calculation of Worst Fuel Unit Capacity

Location	Capacity (worst fuel units)	Basis/Comments
#1 Storage Basin (non HFIR)	2,009	41 rows 49 bundles/row - 9T hardware
#1 Storage Basin (HFIR)	40	Maximum Authorized
#2 Storage Basin (YA-YK)	264	11 rows 24 bundles 9T hardware equivalent
#2 Storage Basin (EBS Racks)	180	6 rows 30 L bundles
Bucket, Bundle, and Test Tube Storage	150	9T hardware equivalent positions
Repackaging Basin	10	
Disassembly Basin	10	
Inspection Basin	10	
Loaded Casks	125	15 casks 50 fuel elements per cask 6 fuel elements per GP tube
Total	2,798	Round up to 2,800

Table 3.3-2 RBOF Basin Radiological Inventory (Ref. 47)

Radionuclide	Activity (Curies) ^{h,k}	Hazard Classification Criteria (Ci)		Ratio to Category 2 Threshold	Hazard Classification
		Category 2 Threshold	Category 3 Threshold		
H-3	1.45E+05	3.0E+05	1.0E+03	4.84E-01	Category 3
Kr-85	2.94E+06	2.8E+07	2.0E+04	1.05E-01	Category 3
Sr-89	1.37E+05	7.7E+05	3.4E+02	1.79E-01	Category 3
Sr-90	2.27E+07	2.2E+04	1.6E+01	1.03E+03	Category 2
Y-90	2.27E+07	4.3E+05 ^g	1.41E+03 ^c	5.27E+01	Category 2
Y-91	5.97E+05	6.5E+05	3.6E+02	9.18E-01	Category 3
Zr-93	1.27E+06	8.9E+04	6.2E+02	1.43E+01	Category 2
Zr-95	2.83E+06	1.5E+06	7.0E+02	1.88E+00	Category 2
Nb-95	1.08E+06	4.3E+05 ^g	9.6E+02 ^c	2.51E+00	Category 2
Nb-95m	9.43E+03	4.3E+05 ^g	5.6E+03 ^a	2.19E-02	Category 3
Tc-99	2.89E+03	3.8E+06	1.7E+03	7.60E-04	Category 3
Ru-103	6.07E+03	4.3E+05 ^g	1.5E+03 ^b	1.41E-02	Category 3
Ru-103m	5.49E+03	4.3E+05 ^g	i	1.28E-02	Category 3
Ru-106	5.91E+07	6.5E+03	1.0E+02	9.09E+03	Category 2
Rh-106	5.91E+07	4.3E+05 ^g	1.96E+03 ^c	1.37E+02	Category 2
Ag-110	6.49E+03	4.3E+05 ^g	i	1.51E-02	Category 3
Ag-110m	4.87E+05	5.3E+05	2.6E+02	9.20E-01	Category 3
Cd-113m	1.95E+04	4.3E+05 ^g	1.1E+01 ^d	4.53E-02	Category 3
Sn-119m	1.10E+04	4.3E+05 ^g	1.8E+03 ^d	2.56E-02	Category 3
Sn-123	4.06E+04	9.5E+05	3.2E+02	4.27E-02	Category 3
Sb-125	2.43E+06	4.3E+05 ^g	1.2E+03 ^b	5.65E+00	Category 2
Te-125m	5.93E+05	4.3E+05 ^g	7.2E+02 ^a	1.38E+00	Category 2
Te-127	2.28E+07	4.3E+05 ^g	1.4E+05 ^a	5.30E+01	Category 2
Te-127m	9.91E+04	1.5E+05	4.0E+02	6.61E-01	Category 3
Te-129	4.48E+00	N/L	2.2E+05 ^e	N/A	Radiological
Te-129m	6.89E+00	1.4E+05	4.0E+02	4.92E-05	Radiological
Cs-134	2.89E+07	6.0E+04	4.2E+01	4.82E+02	Category 2
Cs-137	2.60E+07	8.9E+04	6.0E+01	2.92E+02	Category 2
Ba-137m	2.46E+07	N/L	N/L	N/A	Category 2 ⁱ
Ce-141	2.41E+02	3.3E+06	1.0E+03	7.29E-05	Radiological
Ce-144	1.34E+08	8.2E+04	1.0E+02	1.63E+03	Category 2
Pr-144	1.34E+08	4.3E+05 ^g	1.04E+06 ^c	3.11E+02	Category 2
Pr-144m	1.60E+06	4.3E+05 ^g	i	3.73E+00	Category 2
Pm-147	5.27E+07	8.4E+05	1.0E+03	6.27E+01	Category 2
Pm-148m	3.33E+01	N/L	3.6E+02 ^a	N/A	Radiological

Table 3.3-2 RBOF Basin Radiological Inventory (continued) (Ref. 47)

Radionuclide	Activity (Curies) ^{h,k}	Hazard Classification Criteria (Ci)		Ratio to Category 2 Threshold	Hazard Classification
		Category 2 Threshold	Category 3 Threshold		
Sm-151	1.95E+05	9.9E+05	1.0E+03	1.97E-01	Category 3
Eu-154	2.04E+05	1.1E+05	2.0E+02	1.85E+00	Category 2
Eu-155	1.07E+06	7.3E+05	9.4E+02	1.46E+00	Category 2
Tl-208	2.37E+04	4.3E+05 ^g	i	5.51E-02	Category 3
Pb-209	2.44E+01	4.3E+05 ^g	i	5.68E-05	Category 3
Pb-211	4.65E+01	N/L	6.2E+03 ^d	N/A	Radiological
Pb-212	6.61E+04	4.3E+05 ^g	3.2E+02 ^d	1.54E-01	Category 3
Bi-211	4.65E+01	5.5E+01 ^f	i	8.45E-01	Category 3
Bi-212	6.61E+04	4.3E+05 ^g	2.0E+03 ^d	1.54E-01	Category 3
Bi-213	2.44E+01	N/L	3.2E+03 ^d	N/A	Radiological
Po-212	4.23E+04	5.5E+01 ^f	i	7.84E+02	Category 2
Po-213	2.39E+01	5.5E+01 ^f	i	4.35E-01	Category 3
Po-215	4.65E+01	5.5E+01 ^f	i	8.45E-01	Category 3
Po-216	6.61E+04	5.5E+01 ^f	i	1.20E+03	Category 2
At-217	2.44E+01	5.5E+01 ^f	i	4.44E-01	Category 3
Rn-219	4.65E+01	5.5E+01 ^f	i	8.45E-01	Category 3
Rn-220	6.61E+04	5.5E+01 ^f	2.0E+00 ^d	1.20E+03	Category 2
Fr-221	2.44E+01	5.5E+01 ^f	i	4.44E-01	Category 3
Ra-223	4.65E+01	3.8E+03	6.2E+01	1.22E-02	Radiological
Ra-224	6.61E+04	9.9E+03	2.0E+02	6.67E+00	Category 2
Ra-225	2.44E+01	3.8E+03	7.2E+01	6.43E-03	Radiological
Ac-225	2.44E+01	2.9E+03	3.2E+01	8.43E-03	Radiological
Ac-227	4.79E+01	4.3E+00	4.2E-02	1.11E+01	Category 2
Th-227	4.59E+01	5.5E+01 ^f	3.2E+01	8.35E-01	Category 3
Th-228	6.58E+04	9.2E+01	1.0E+00	7.15E+02	Category 2
Th-229	2.44E+01	5.5E+01 ^f	i	4.44E-01	Category 3
Th-231	3.19E+01	5.5E+01 ^f	i	5.81E-01	Category 3
Th-232	4.81E+01	1.8E+01	1.0E-01	2.67E+00	Category 2
Th-234	6.05E-01	N/L	2.8E+03 ^c	N/A	Radiological
Pa-231	6.39E+02	5.5E+01 ^f	2.0E-01 ^d	1.16E+01	Category 2
Pa-233	2.41E+02	N/L	4.6E+03 ^d	N/A	Radiological
Pa-234m	6.05E-01	4.3E+05 ^g	N/L	1.41E-06	Category 3
U-232	1.15E+05	5.5E+01 ^f	8.2E-01 ^a	2.08E+03	Category 2
U-233	1.10E+05	2.2E+02	4.2E+00	5.00E+02	Category 2
U-234	5.37E+03	2.2E+02	4.2E+00	2.44E+01	Category 2
U-235	3.19E+01	2.4E+02	4.2E+00	1.33E-01	Category 3

Table 3.3-2 RBOF Basin Radiological Inventory (continued) (Ref. 47)

Radionuclide	Activity (Curies) ^{h,k}	Hazard Classification Criteria (Ci)		Ratio to Category 2 Threshold	Hazard Classification
		Category 2 Threshold	Category 3 Threshold		
U-236	9.21E+01	5.5E+01 ^f	4.2E+00 ^b	1.68E+00	Category 2
U-237	7.25E+02	N/L	1.4E+04 ^d	N/A	Radiological
U-238	2.35E+02	2.4E+02	4.2E+00	9.81E-01	Category 3
Np-237	2.47E+01	5.8E+01	4.2E-01	4.26E-01	Category 3
Np-239	2.70E+04	4.3E+05	7.8E+03 ^d	6.27E-02	Category 3
Pu-236	3.13E+05	5.5E+01 ^f	2.0E+00 ^a	5.69E+03	Category 2
Pu-238	1.45E+05	6.2E+01	6.2E-01	2.34E+03	Category 2
Pu-239	1.63E+05	5.6E+01	5.2E-01	2.91E+03	Category 2
Pu-240	2.74E+07	5.5E+01 ^f	5.2E-01 ^c	4.98E+05	Category 2
Pu-241	2.97E+07	2.9E+03	3.2E+01	1.02E+04	Category 2
Am-241	1.45E+05	5.5E+01	5.2E-01	2.64E+03	Category 2
Am-242m	9.55E+02	5.6E+01	5.2E-01	1.70E+01	Category 2
Am-242	9.52E+02	N/L	8.2E+03 ^d	N/A	Radiological
Am-243	2.70E+04	5.5E+01	5.2E-01	4.91E+02	Category 2
Cm-242	1.37E+06	1.7E+03	3.2E+01 ^c	8.09E+02	Category 2
Cm-243	1.37E+04	5.5E+01 ^f	8.2E-01	2.50E+02	Category 2
Cm-244	7.70E+06	5.5E+01 ^f	1.04E+00	1.40E+05	Category 2
Cm-246	6.02E+02	5.5E+01 ^f	5.2E-01 ^c	1.09E+01	Category 2
Grand Total	6.69E+08		Total ratio =	6.83E+05	

N/A Not Available

N/L Not listed in DOE-STD-1027-92.

a Memorandum SRT-SSA-930073, DOE-STD-1027-92 Category 3 Threshold Limits, November 22, 1993.

b Memorandum SRT-SSA-930011, DOE-STD-1027-92 Category 3 Threshold Limits, October 19, 1993.

c Memorandum SRT-LSA-93-0087, DOE-STD-1027-92 Category 3 Threshold Limits, June 14, 1993.

d Memorandum EPD-TAF-94-0008, DOE-STD-1027-92 Category 3 Threshold Limits, June 7, 1994.

e Memorandum SRT-SSA-940048, DOE-STD-1027-92 Category 3 Threshold Limits, February 18, 1994.

f Based on DOE-STD-1027-92 alpha emitter standard.

g Based on DOE-STD-1027-92 beta-gamma emitter standard.

h Individual radionuclide contents x 2,800 RFAs from SFS-RPS-970053 (Ref. 221).

Table 3.3-2 RBOF Basin Radiological Inventory (continued) (Ref. 47)

- i These radioisotopes have extremely short half-lives (< 10 min.) and a Hazard Category 3 threshold limit is difficult to determine. The isotopes are believed to be daughter isotopes. Therefore, the isotopes are compared to the Hazard Category 2 threshold limits for alpha and beta-gamma emitters according to DOE-STD-1027-92 and categorized appropriately.
- j Ba-137m is not listed as either an alpha or a beta-gamma emitter on the chart of nuclides, so it is conservatively assumed to be a Category 2 hazard, even though its half-life is 2.5 minutes.
- k Table 2 of N-CLC-H-00084 (Ref. 21).

Table 3.3-3 RBOF Hazardous Chemical Inventory

Hazardous Chemical	Maximum Inventory	Reactivity Group Name	RGN ^b	Location
Sodium hydroxide (50%)	1,100 gallons (tank)	Caustics	10	Chemical Pad No. 1
Nitric acid (50%)	1,100 gallons (tank)	Acids, mineral oxidizing	2	Chemical Pad No. 1
Sodium nitrite (30%)	900 gallons	Oxidizing agents, strong	104	Chemical Pad No. 2
Aluminum nitrate (34.1%)	No Longer Used	Oxidizing agents, strong	104	Not Available
Azo violet ^a	12.5 grams	Azo compounds	8	Building 244-1H
Boric acid ^a	1,100 pounds	Acids, mineral nonoxidizing	1	Building 244-1H
Celite 545 ^a	5,000 pounds	N/A (nonreactive)	N/A	Building 244-1H
Anion ion-exchange resin ^a	420 cubic feet	N/A (nonreactive)	N/A	Building 244-1H
Cation ion-exchange resin ^a	300 cubic feet	N/A (nonreactive, except when exposed to concentrated nitric acid)	N/A	Building 244-1H
Kemp K-11 vapoilsorb (activated bauxite) ^a	250 pounds	N/A (insoluble in water)	N/A	Building 244-1H
Kemp K-25 (alumina hydrate) ^a	250 pounds	N/A (insoluble in water)	N/A	Building 244-1H
Oxalic acid, anhydrous ^a	525 pounds	Acids, organic	3	Building 244-1H
Potassium tetraborate ^a	100 pounds	Metals, alkali and alkali earth, elemental	21	Building 244-1H
Soda ash (sodium carbonate) ^a	100 pounds	Caustics	10	Building 244-H
Sodium dichromate, anhydrous ^a	1,000 pounds	Metals and metal compounds, toxic / oxidizing agents, strong	24 / 104	Building 244-1H
67-100 Steam cleaner (sodium metasilicate/ sodium hydroxide) ^a	1,000 pounds	Caustics	10	Building 244-H
Ethylene Glycol	310 gallons	Alcohols and Glycols	4	HVAC Chiller System

a Dry chemicals stored in manufacturers' packaging.

b Reactivity group number from the Handbook of Chemical Hazard Analysis Procedures (Ref. 26).

N/A Not Applicable

Table 3.3-4 RBOF Hazardous Chemical Inventory and Comparison to Limits

Chemical	Inventory (lb)	Reportable Quantity ^a (lb)	Threshold Planning Quantity ^b (lb)	Threshold Quantity ^c (lb)	RBOF Inventory Limit
34.1% Aluminum Nitrate	208	NL	NL	NL	None
Azo Violet	0.03	NL	NL	NL	None
4-(p-nitrophenyl)azolresorcinol					
magneson					
Boric Acid	100	NL	NL	NL	100 pounds
Celite 545	2,700				None
flux calcined diatomaceous earth		NL	NL	NL	
crystalline silica		NL	NL	NL	
cristobalite quartz		NL	NL	NL	
Ethylene Glycol	3,855	NL	NL	NL	3,855 pounds
Ion Exchange Resin (anion form)	40 ft ³	NL	NL	NL	None
Ion Exchange Resin (cation form)	30 ft ³	NL	NL	NL	None
Kemp K-11 Vapoilsorb activated bauxite	250	NL	NL	NL	None
Kemp K-25 Grade A Alumina (alumina hydrate)	250	NL	NL	NL	None
50% Nitric Acid	5,986 ^e	1,000	1,000	NL	5,986 pounds
Oxalic Acid Anhydrous	525	NL	NL	NL	525 pounds
75% Phosphoric Acid	<5,000	5,000	NL	NL	<5,000 pounds
Potassium Tetraborate	100	NL	NL	NL	None
Soda Ash (sodium carbonate)	100	NL	NL	NL	100 pounds
Sodium Dichromate Anhydrous	1,000	NL	NL	NL	1,000 pounds
50% Sodium Hydroxide	6,919 ^e	1,000	NL	NL	6,919 pounds
30% Sodium Nitrite ^d	3,043 ^e	100	NL	NL	3,043 pounds
Steam Cleaner 67-100					
33% sodium metasilicate	334	NL	NL	NL	None
5% sodium hydroxide	50	1,000	NL	NL	1,000 pounds

NL Not Listed in the associated CFR

a Reportable Quantity as given in 40 CFR 302.4 (Ref. 29)

b Threshold Planning Quantity as given in 40 CFR 355 (Ref. 29)

c Threshold Quantity as given in 29 CFR 1910.119 (Ref. 30)

d Chemical inventory controlled due to concerns with mixing of nitric acid

e Inventory (lb) = gal x specific gravity x 8.345 lb/gal x %concentration (See Reference 13 for calculations)

"None" implies that there is no limit to the amount to be stored.

Table 3.3-5 RBOF/RRF Hazardous Energy Source List

Group	Initiator	Examples		
Potential	Electrical	High voltage Transformers Wiring Switchgear	Underground wiring Cable runs Service outlets, fittings Pumps	Motors Heaters Power tools Small equipment
Gravitational		Ladders Suspended objects Human effort Stairs Lifts Cranes Trucks	Pits Elevated doors Canals Vessels Slings Hoists Scaffolds and ladders	Falling objects Traveling bridge Monorail system Turntables Trolleys Chainfalls Water basins
Potential	Pressure	Gas bottles, Air receiver tank	Heated storage tanks	Stressed members
Kinetic	Linear	Railroad cars Trucks Dollies Power-driving tools	Railroad Surfaces, edges, pinch points	Obstruction Presses Crane loads in motion
Kinetic	Rotational	Motors Pumps Fans	Gears Grinders, saws, brushes, etc.	Pulleys, belts, etc.
Radiation	Ionizing	Casks Canals and basins Dollies Trucks Irradiated equipment	Hand carry Cranes, lifts Storage, receiving, shipping	Shops, hot cells Assembly/inspection Radioactive sources Contamination
Radiation	Nonionizing	Electromagnetic: Welding	Thermal: Steam lines	Acoustical: Equipment noise
Other	Thermal	Thermal: Convection Welding Exposed steam pipes Electric heaters Electric equipment	Hot casks > 120°F	

Table 3.3-6 RBOF/RRF Hazardous Material Source List

Groups	Examples
Radioactive	All radionuclides
Fissionable	U-233, U-235, U-238, Pu-238, Pu-239, Pu-240, Pu-241
Alkali metal	NaK
Toxic	Heavy metals, quartz, acid, diatomaceous earth, nitrophenyl, ethylene glycol
Carcinogen	Sodium nitrite
Irritant	Acid, caustics
Asphyxiant	Nitrogen, hydrogen, low oxygen areas (pits, tunnels), nitric acid
Corrosive	Acids, caustics
Explosive	Hydrogen, sodium nitrite (when heated to 537 °C or in contact with reducing materials)
Combustible/flammable	Paper, rags, boxes, walls, insulation, wood, gases (e.g., hydrogen, methane, ethane)
Fire promotion (increases flammability of combustible materials)	Nitric acid (when in contact with organic materials)
Oxidizing	Dichromate, nitric acid, aluminum nitrate, sodium nitrates

Table 3.3-7 Qualitative Frequency Classification Guidelines (Ref. 2)

Code	Description	Estimated Annual Probability of Occurrence
A = Anticipated	Incidents that may occur several times during the lifetime of the facility. (Incidents that commonly occur.)	$p > 10^{-2}$
U = Unlikely	Accidents that are not anticipated to occur during the lifetime of the facility. Natural phenomena of this probability class include Uniform Building Code level earthquake, 100-year flood, and maximum wind gust.	$10^{-2} > p > 10^{-4}$
EU = Extremely Unlikely	Accidents that will probably not occur during the life cycle of the facility. This class includes DBAs.	$10^{-4} > p > 10^{-6}$
BEU = Beyond Extremely Unlikely	All other accidents.	$10^{-6} > p$

Table 3.3-8 Radiological Accident Consequence Level Guidelines

Consequence Level Description ^a	Public (MOI) ^b	Facility Worker (MEI) ^c
High	0.5 rem at site boundary	Offsite consequence is bounding.
Moderate	N/A	> 25 rem or life-threatening injury (i.e., loss of life, limb, or eye).
Low	N/A	< 25 rem or nonlife-threatening injury (i.e., no permanent disabilities). Requires hospitalization.
Negligible	N/A	< 0.5 rem or only a minor injury.

a The consequence level is defined as follows (Ref. 2, 56):

High: Considerable onsite and offsite impact on people or the environs.

Moderate: Considerable onsite impact on people or the environs; only minor offsite impact.

Low: Minor onsite and negligible offsite impact on people or the environs.

Negligible: Negligible onsite and offsite impact on people or the environs.

b MOI - Maximally Exposed Offsite Individual

c MEI - Maximally Exposed Individual

Table 3.3-9 Chemical Accident Consequence Level Guidelines

Consequence Level	Public (MOI)	Facility Worker
High	PEL-TWA 2E-06 ICR	Offsite Consequence is bounding.
Moderate	N/A	> ERPG-2 or life-threatening injury (i.e., loss of life, limb, or eye).
Low	N/A	< ERPG-2 or nonlife-threatening injury (i.e., no permanent disabilities). Requires hospitalization.
Negligible	N/A	< ERPG-1 or only a minor injury.

PEL - Permissible Exposure Limit

TWA - Time-Weighted Average

ICR - Incremental Cancer Risk

N/A - Not available

Table 3.3-10 Risk Matrix: Consequence Versus Frequency Guidelines

Frequency Level	Consequence Level			
	Negligible	Low	Moderate	High
Anticipated $p > 10^{-2}$	10	7	4	1
Unlikely $10^{-2} \geq p > 10^{-4}$	11	8	5	2
Extremely Unlikely $10^{-4} \geq p \geq 10^{-6}$	12	9	6	3
Beyond Extremely Unlikely $10^{-6} < p$	13	13	13	13

The consequence level is defined as follows:

- High: Considerable onsite and offsite impact on people or the environs.
- Moderate: Considerable onsite impact on people or the environs; only minor offsite impact.
- Low: Minor onsite and negligible offsite impact on people or the environs.
- Negligible: Negligible onsite and offsite impact on people or the environs.

Combinations of conclusions from hazard evaluation that requires a bounding analysis for each event type.

Combinations of conclusions from hazard evaluation that requires quantitative analysis for the identified event.

Table 3.3-11 Risk Rank Binning Summary

Event Category	Risk Rank													Standard Industrial Hazards
	1	2	3	4	5	6	7	8	9	10	11	12	13	
E-1				310		400, 400b, 313, 314	3, 400a	2a						1, 2, 3a, 311, 312, 450, 451, 501
E-2				314a, 316, 318			4, 5, 5a, 6, 7, 7a, 8b, 10, 10a, 11, 11a, 12, 13, 314b, 315, 317, 328c, 402	8c		9, 321, 324, 401	8		8a, 401a	
E-3								16, 17, 18	19, 329c	21	20			
E-4							14, 22, 23, 25, 334							26, 26a, 27, 28, 29, 30, 31, 32, 33, 33a, 33b, 330, 331, 332, 333, 335
E-5			43		34, 35, 37b, 38, 39, 40, 40a, 41, 42a, 44	36, 37a							24, 40b, 42, 404	
E-6				51						15, 48, 49, 53, 54, 55, 319				45, 46, 47, 56, 337, 338, 339, 340
E-7										57, 58, 342, 406, 407	341			
E-8							59, 59a			52				60, 61, 343
E-9					345c, 345d			345a					62, 345	
E-10			63a		63, 64, 347, 348	65		347a	66	69	349		67, 68, 346	
Quantity	----	----	2	5	16	7	27	7	3	19	4	---	11	35

Table 3.3-12 Distribution of Events By WSRC-TR-94-0466 Categories (Ref. 56)

Risk Rank													
Event Category *	1	2	3	4	5	6	7	8	9	10	11	12	13
Natural Phenomena Events													
Earthquake			43, 63a	314a, 316, 318	42a, 63, 347		402	18, 347a		321, 401	8, 20		401a
Winds			63a	314a, 316, 318	42a, 63, 64, 347	65	402	18, 347a	19, 329c	321, 401	20		401a
Lightning										69	349		
Temp. Extremes					348								67
Rain									66				
Meteorite													68, 346
Volcano, Flood, and Flora/Fauna	Not applicable to RBOF												
Externally Induced Events													
Surface Vehicle Impact				314a, 316, 318	42a, 345c, 345d	313	5, 5a, 402	16, 345a					345
Aircraft													62, 345
Hoist, Crane, Rigging Failure					37b, 42a	36, 37a	5, 5a, 8c						
Structural Failure			43		42a		4, 23, 402			21			
Adjacent Facility Accidents						313				58, 342, 406, 407			
Missiles						65	402		19, 329c	321, 401	20		
Process-Related Events													
Nuclear Criticality			43		34, 35, 37b, 38, 39, 40, 40a, 41, 42a, 44	36, 37a, 64, 65							24, 40b, 42, 404
Fire				310, 501		313, 400, 400 b	3, 451a	2a					
Uncont. Chem. Reactions				51		400	400a						
Air Reversal	Not applicable to RBOF												
Overflow				314a, 316, 318			314b, 315, 317, 328c			9, 321, 326b			
Spills					40, 40a		59, 59a, 334			401			
Leaks				314a, 316, 318			6, 7, 7a, 8b, 10, 10a, 11, 11a, 12, 13, 59a, 314b, 315, 317, 328c, 334	17		15, 21, 48, 49, 319, 321, 324, 401, 406, 407			8a
Explosions				51		314, 400	3, 400a						
Other							14, 22, 25, 59			52, 53, 54, 55, 57, 58, 338, 339	341		

* Does not include standard industrial accidents as identified in the event evaluation tables. Some events may fall into more than one category.

Table 3.3-13 Hazard Evaluation - RBOF Basin Area

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Preventive features		Freq. Level & Source	Method of detection	Mitigative features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
1	E-1	Ignition of combustibles resulting in area or equipment fire	Welding spark, electrical short, unknown ignition source, worker error, improper maintenance operation	None	Fire Protection Program (Restriction of ignition sources and combustible materials)	A 1	Smell and Visual (Fire detectors are not installed)	Facility sections separated by partitions and shield walls, fire extinguishers, emergency exits	<u>Fire Protection Program</u> , <u>Event Response Procedures</u>	Shutdown of affected area operations, water damage to other equipment in the area, property damage	Low consequence. Standard Industrial Hazard.	7	
2	E-1	Tractor-trailer fire while in Carport (ignition of combustibles and flammables)	Electrical short, unknown ignition source	Standard Vehicle design	<u>Fire Protection Program</u> (limits combustibles and Time tractor is in Carport is kept to a minimum)	U 3	Smell and Visual (Fire detectors are not installed)	Facility sections separated by partitions and shield walls, fire extinguishers, emergency exits	<u>Fire Protection Program</u> , <u>Event Response Procedures</u>	Shutdown of affected area operations, water damage to other equipment in the area, property damage	Low consequence. Standard Industrial Hazard.	8	
2a	E-1	Tractor-trailer fire while in Carport (ignition of combustibles and flammables) engulfs cask	Electrical short, unknown ignition source	Standard Vehicle design	<u>Fire Protection Program</u> (limits combustibles and Time tractor is in Carport is kept to a minimum)	U 3	Smell and Visual (Fire detectors are not installed)	Facility sections separated by partitions and shield walls, fire extinguishers, emergency exits. <u>Cask Design</u>	<u>Event Response Procedures</u> , <u>Fire Protection Program</u> , <u>Fuel Receipt & Shipping Program</u>	Shutdown of affected area operations, water damage to other equipment in the area, property damage	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	8	

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence												
Event No	Event Category	Postulated Event Description	Causes	Prevention features		Freq Level & Source	Method of detection	Mitigation features			Consequence level	Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems		
3	E-1	Minor explosion during venting (through building exhaust system) of offsite cask or onsite cask	Ignition of explosive mixture of hydrogen, oxygen, and air in cask due to radiolysis or uranium metal-oxygen reaction	Cask design, "Once through" air-flow rate (Negative pressure in basin area)	Water removal before shipment of cask, use of ignition-free vacuum, inert gas blanket, Fuel Receipt and Shipping Program (control flammable gas generation)	A 1	Visual observation audible	Cask and equipment design, ventilation system, RBOF's containment capability	Event Response Procedures, Fire Protection Program	Stoppage of operations in Cask Wash Pit and or Cask Unloading Basin, property damage	Low consequence ≤ 25 rem or non-life threatening injury to the onsite worker.	7
3a	E-1	Lightning strikes facility resulting in fire	Natural phenomena	Standard building design (prevents significant lightning damage)	RBOF/RRF Building Design	A 2	Audible, visual	Standard building design	Event Response Procedures, Fire Protection Program	Property damage	Low consequence Standard Industrial Hazard	7

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes.

- (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Postulated event occurrence		
				Design	Administrative			Design	Administrative	Consequence level
4	E-2	Breach of Shipping Cask noted during radiation survey or receipt inspection on arrival in Carport	Material failure, over pressurization, thermal stress, mechanical damage	Cask design	Fuel Receipt and Shipping Program (Receipt of casks with known fuel damage requires approval before shipment to RBOF), DOT and NRC regulations	A 1	HP surveys, visual indications	RBOF Confinement capability, Radiation Monitoring, Cask Design	Radiological Protection Program, Fuel Receipt and Shipping Program	Impact on other systems Health and Safety Low consequence, ≤ 25 rem or non-life threatening injury to the onsite worker.
5	E-2	Airborne release occurs when offsite cask is dropped or impacted by vehicle	Equipment failure, inadequate procedures, worker error	Cask Design, design of lifting devices and fixtures	Hoisting and Rigging Program, NOPs	A 3	Visual and Audible Radiation Monitoring	Radiation Monitoring, Cask Design	Damage and or contamination of area equipment resulting in shutdown of affected area operations	Low consequence, ≤ 25 rem or non-life threatening injury to the onsite worker.
5a	E-2	Airborne release occurs when onsite cask is dropped or impacted by vehicle	Equipment failure, inadequate procedures, worker error	Cask Design, design of lifting devices and fixtures	Hoisting and Rigging Program, NOPs, Fuel Receipt and Shipping Program (inventory control - no known leaking fuel loaded into cask)	A 3	Visual and Audible Radiation Monitoring	Radiation Monitoring, Cask Design	Damage and or contamination of area equipment resulting in shutdown of affected area operations	Low consequence, ≤ 25 rem or non-life threatening injury to the onsite worker.

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Postulated event occurrence			Risk Rank
				Design	Administrative			Design	Administrative	Consequence level	
6	E-2	Airborne release to RBOF occurs during cask venting, through the building exhaust system, prior to transfer from Cask Wash Pit to Cask Basin	Equipment failure, power interruption, inadequate procedure, worker error,	"Once-through" airflow with basin areas at a lower pressure than the rest of the building, HP survey instruments provided in key locations, <u>cask design</u>	NOPs, preventive maintenance, Fuel Receipt and Shipping Program (inventory control - no known leaking fuel loaded into cask)	A 1	Visual and Audible Radiation Monitoring	Radiation Monitoring, <u>Cask Design</u>	<u>Radiological Protection Program</u> (cask venting samples), <u>Event Response Procedures</u> , <u>Fuel Receipt and Shipping Program</u> (inventory control)	Shutdown of affected area operations Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	7
7	E-2	Airborne release occurs when fuel ruptured during handling or movement	Equipment failure, inadequate procedures, worker error	Design of lifting/moving devices and fixtures comply with design requirements, safety factors used in equipment design	Nuclear Criticality Safety Program (fuel handling procedures), Hoisting and Rigging Program	A 1	Visual and Audible Radiation Monitoring	Remote operations, basin water shielding, building ventilation and exhaust systems, RBOF confinement, basin water purification system, Radiation Monitoring	<u>Fuel Receipt & Shipping Program</u> [inventory control (1050 Ci of Kr-85), 2 year cooling (no other noble gases), and total inventory (2800 RFAs)], <u>Cask/Fuel Receiving Requirements</u> , <u>Radiological Protection Program</u> , <u>Event Response Procedures</u>	Increase basin water contamination and loading of radionuclides on purification filters Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	7

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence												
		Prevention features			Freq. Level & Source	Method of detection	Mitigation features			Consequence level		
Event No.	Event Category	Postulated Event Description	Causes	Design			Administrative	Design	Administrative	Impact on other systems	Health and Safety	Risk Rank
7a	E-2	Target Assemblies ruptured during handling or movement	Equipment failure, inadequate procedures, worker error	Design of lifting/moving devices and fixtures comply with design requirements, safety factors used in equipment design	<u>Nuclear Criticality Safety Program</u> (control of fuel cropping operations), <u>Hoisting and Rigging Program</u>	A 1	Visual and Audible Radiation Monitoring	Remote operations, basin water shielding, building ventilation and exhaust systems, RBOF confinement, basin water purification system, Radiation Monitoring	<u>Fuel Receipt & Shipping Program</u> (Mark VI-B tritium inventory controlled - 65 Ci), <u>Cask/Fuel Receiving Requirements</u> , <u>Radiological Protection Program</u> , <u>Event Response Procedures</u>	Increase basin water contamination and loading of radionuclides on purification filters	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	7
8	E-2	Earthquake damage or dropped shipping cask (crack in basin wall) causes basin draindown and loss of contaminated basin water to subsoil (environment)	Natural Phenomena, equipment failure, inadequate procedures, worker error	<u>Basin Design</u>	<u>Hoisting and Rigging Program</u>	U 4	Basin level controllers, noticeable ground movement, Visual and Audible Radiation Monitoring	<u>Basin Design</u>	<u>Basin Water Chemistry Control Program</u> , <u>Event Response Procedures</u> , <u>Radiological Protection Program</u> , water level checks	Demand for water makeup, increases in area radiation levels resulting in shutdown of area operations, potential for major property damage	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	11

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
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Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
8a	E-2	Low water level in basin allows decay heat to deform fuel	Failure to add water to basin due to leak in water supply line or failure of level controller	<u>Rack, Bundles, Buckets, Bucket Inserts, Blocking Devices, and Hardware Design,</u>	<u>Fuel Receipt and Shipping Program (maximum heat load), Cask/Fuel Receiving Requirements, Nuclear Criticality Safety Program, Criticality Control Requirements (store fuel properly),</u>	BEU 2	Visual and Audible Radiation Monitoring	Shielding capability of RBOF structure and basin water, rack configuration, ventilation and exhaust system, NIMs and Remote Alarm	<u>Fuel Receipt and Shipping Program (maximum heat load), Cask/Fuel Receiving Requirements, water level checks, Event Response Procedures, Radiological Protection Program</u>	Additional activity in basin water will be removed by basin purification filter. Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	High consequence. ≥ 0.5 rem at site boundary.	13	
8b	E-2	Low water level in basin due to FD system leak with fuel exposed	Equipment failure, inadequate procedure, improper maintenance, worker error	Basin Penetration Design	<u>Fuel Receipt and Shipping Program (maximum heat load), Cask/Fuel Receiving Requirements, Nuclear Criticality Safety Program, Criticality Control Requirements (store fuel properly),</u>	A 2	Visual and Audible Radiation Monitoring	Shielding capability of RBOF structure and basin water, rack configuration, ventilation and exhaust system, NIMs and Remote Alarm	<u>Basin Water Chemistry Control Program, Fuel Receipt and Shipping Program (maximum heat load), Cask/Fuel Receiving Requirements, Event Response Procedures, Radiological Protection Program</u>	Additional activity in basin water will be removed by basin purification filter. Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	7	

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
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Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
8c	E-2	Low water level due to basin leak with high radiation due to exposed fuel and airborne release due to resuspension of basin water activity	Dropped shipping cask	Basin Design, Rack, Bundles, Buckets, Bucket Inserts, Blocking Devices, and Hardware Design.	Fuel Receipt and Shipping Program (maximum heat load), Hoisting and Rigging Program, Cask/Fuel Receiving Requirements, Nuclear Criticality Safety Program, Criticality Control Requirements (store fuel properly).	U 1	Visual and Audible Radiation Monitoring	Shielding capability of RBOF structure and basin water, rack configuration, NIMs	Basin Water Chemistry Control Program, Fuel Receipt & Shipping Program (maximum heat load), Cask/Fuel Receiving Requirements, Radiological Protection Program, Event Response Procedures	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	8	
9	E-2	Basins overflow	Equipment failure, inadequate procedure, improper maintenance, worker error	Basin Level Control System, Interlocks	NOPs, routine preventive maintenance	A 1	Visual, Basin level indicators	RBOF confinement capability, basin level control and level monitoring system	Basin Water Chemistry Control Program (maintain activity of all radionuclides), Event Response Procedures, Radiological Protection Program, Spill Response Procedures	Contamination of uncontrolled areas	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	10	

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
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Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
10	E-2	Airborne release occurs from inadvertent cutting of fuel during cropping in the disassembly basin	Equipment malfunction, inadequate procedures, worker error, error on blue prints	Design of cropping equipment complies (saw table stops, measurement rods)	Nuclear Criticality Safety Program (control of fuel cropping operations).	A 1	Visual and Audible Radiation Monitoring	Remote cropping operations, basin water shielding, "Once-through" airflow and exhaust through building exhaust filters, Radiation Monitoring	<u>Fuel Receipt & Shipping Program</u> [Inventory control (1050 Ci of Kr-85), 2 year cooling (no other noble gases), and total inventory (2800 RFAs)], <u>Cask/Fuel Receiving Requirements</u> , <u>Radiological Protection Program</u> , <u>Event Response Procedures</u>	Basin water contamination, high airborne activity resulting in shutdown of affected area, increased radionuclides loading on purification filter	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	7	
10a	E-2	Airborne release occurs from inadvertent cutting of target core during cropping in the disassembly basin	Equipment malfunction, inadequate procedures, worker error, error on blue prints	Design of cropping equipment (saw table stops, measurement rods)	Nuclear Criticality Safety Program (control of fuel cropping operations)	A 1	Visual and Audible Radiation Monitoring	Remote cropping operations, basin water shielding, "Once-through" airflow and exhaust through building exhaust filters, Radiation Monitoring	<u>Fuel Receipt & Shipping Program</u> (Mark VI-B tritium inventory controlled - 65 Ci), <u>Cask/Fuel Receiving Requirements</u> , <u>Radiological Protection Program</u> , <u>Event Response Procedures</u>	Basin water contamination, high airborne activity resulting in shutdown of affected area, increased radionuclides loading on purification filter	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	7	

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
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Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
11	E-2	Airborne release occurs when fuel ruptures while in storage basins	Material failure, thermal stress, corrosion, chemical reaction, mechanical damage	Fuel/Target assembly and storage rack design, evaporative cooling, basin water level control system, compatible materials	Fuel Receipt & Shipping Program (inventory control (1050 Ci of Kr-85), 2 year cooling (no other noble gases), and total inventory (2800 RFAs)), Basin Water Chemistry Control Program	A 1	Visual and Audible Radiation Monitoring	Basin water shielding, building "Once-through" airflow and exhaust through building exhaust filters, RBOF confinement capability, Radiation Monitoring	<u>Fuel Receipt & Shipping Program</u> (inventory control (1050 Ci of Kr-85), 2 year cooling (no other noble gases), and total inventory (2800 RFAs)), <u>Cask/Fuel Receiving Requirements, Radiological Protection Program, Event Response Procedures</u>	Basin water contamination, high airborne activity resulting in shutdown of affected area, increased radionuclide loading on purification filter	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	7	
11a	E-2	Airborne release occurs when Target Assemblies rupture while in storage basins	Material failure, thermal stress, corrosion, chemical reaction, mechanical damage	Fuel/Target assembly and storage rack design, evaporative cooling, basin water level control system, compatible materials	Basin Water Chemistry Control Program	A 1	Visual and Audible Radiation Monitoring	Basin water shielding, building "Once-through" airflow and exhaust through building exhaust filters, RBOF confinement capability, Radiation Monitoring	<u>Fuel Receipt & Shipping Program</u> (Mark VI-B tritium inventory controlled - 65 Ci), <u>Cask/Fuel Receiving Requirements, Radiological Protection Program, Event Response Procedures</u>	Basin water contamination, high airborne activity resulting in shutdown of affected area, increased radionuclide loading on purification filter	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	7	

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Postulated event occurrence			Risk Rank
				Design	Administrative			Design	Administrative	Consequence level	
12	E-2	Airborne release to RBOF occurs during cask cooling through the Building Exhaust system	Bad connection between vessel vent hose and cask vent valve assembly, leak in vessel vent hose, worker error	"Once-through" airflow with basin areas at a lower pressure than the rest of the building, HP survey instruments provided in key locations, <u>cask design</u>	NOPs, preventive maintenance, Fuel Receipt and Shipping Program (inventory control - no known leaking fuel loaded into cask).	A 1	Visual and Audible Radiation Monitoring	Radiation Monitoring, <u>Cask Design</u>	<u>Radiological Protection Program</u> (cask venting samples), <u>Event Response Procedures</u> , <u>Fuel Receipt and Shipping Program</u> (inventory control)	Shutdown of affected area operations Low consequence, ≤ 25 rem or non-life threatening injury to the onsite worker.	7
13	E-2	Airborne release to RBOF occurs during cask cooling between cask and desuperheater	Bad connection, leak in hose, worker error, bad valve	"Once-through" airflow with basin areas at a lower pressure than the rest of the building, HP survey instruments provided in key locations, <u>cask design</u>	NOPs, preventive maintenance, Fuel Receipt and Shipping Program (inventory control - no known leaking fuel loaded into cask)	A 1	Visual and Audible Radiation Monitoring	Radiation Monitoring, <u>Cask Design</u>	<u>Radiological Protection Program</u> (cask venting samples), <u>Event Response Procedures</u> , <u>Fuel Receipt and Shipping Program</u> (inventory control)	Shutdown of affected area operations Low consequence, ≤ 25 rem or non-life threatening injury to the onsite worker.	7
14	E-4	Excessive radiation exposure occurs when fuel is unloaded in air	Procedure violation, worker error	None	NOPs, constant monitoring of radiation levels during air unloading operations.	A 1	Worker observation, Visual and Audible Radiation Monitoring	RBOF Facility functions as a confinement system, Radiation Monitoring	<u>Radiological Protection Program</u> , <u>Event Response Procedures</u>	Shutdown of affected area operations Low consequence, ≤ 25 rem or non-life threatening injury to the onsite worker.	7

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
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Postulated event occurrence												
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Risk Rank	
				Design	Administrative			Design	Administrative	Impact on other systems		Health and Safety
15	E-6	Loss of nitrogen during cask cooling between source and cask	Bad connection, source to RBOF out, leaking valve or hose, worker error	Engineering Standards	Preventive maintenance, routine testing	A I	Flow and pressure meters, Visual and Audible Radiation Monitoring	None	Normal shutdown of all affected operations	Delay in cooling of cask	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	10
16	E-3	Unfiltered ground release occurs when Carport Door is damaged by tractor trailer or train while delivering fuel	Equipment failure, inadequate procedures, worker error	Train tracks, RBOF and door construction and location, SRS vehicle control, SRS security fences and gates	8Q Safety Program (spotter during backup of trailer)	U I	Visual and Audible Radiation Monitoring	Radiation Monitoring	<u>Radiological Protection Program</u> , <u>Event Response Procedures</u> , <u>Fuel Receipt & Shipping Program</u> , <u>Cask/Fuel Receiving Requirements</u>	Compromises the confinement capability of RBOF Facility, property damage, downtime	Low consequence, ≤ 25 rem or non-life threatening injury to the onsite worker.	8
17	E-3	Unfiltered ground release through carport door when door is open	Mechanical failure, equipment failure	Backup manual controls and equipment	Manual operation procedure, routine testing.	U I	Visual and Audible Radiation Monitoring	Building ventilation system, Radiation Monitoring	<u>Radiological Protection Program</u> , <u>Event Response Procedures</u> , <u>Fuel Receipt & Shipping Program</u> , <u>Cask/Fuel Receiving Requirements</u>	Could limit facility operations	Low consequence, ≤ 25 rem or non-life threatening injury to the onsite worker.	8

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
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Postulated event occurrence												
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level	
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety	
18	E-3	Unfiltered ground release occurs when an earthquake or high winds damages RBOF structurally resulting in fuel damage	Natural Phenomena	<u>RBOF Building Design, Rack Bundles, Buckets, Bucket Inserts, Blocking Devices, and Hardware Design.</u>	<u>Criticality Control Requirements (store fuel properly), Nuclear Criticality Safety Program, NSDS</u>	U 4	Visual and Audible Radiation Monitoring	RBOF structure, shield walls, basins below grade, basin water, basin (including inspection basin door) and canal system, Radiation Monitoring	<u>Radiological Protection Program, Event Response Procedures, Fuel Receipt & Shipping Program, Cask/Fuel Receiving Requirements</u>	Compromises the confinement capability of RBOF Facility, property damage	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	8
19	E-3	Unfiltered ground release occurs when a tornado or missile damages RBOF structurally resulting in fuel damage	Natural Phenomena	<u>Rack, Bundles, Buckets, Bucket Inserts, Blocking Devices, and Hardware Design.</u>	<u>Criticality Control Requirements (store fuel properly), Nuclear Criticality Safety Program, NSDS</u>	EU 4	Visual and Audible Radiation Monitoring	RBOF structure, shield walls, basins below grade, basin water, basin (including inspection basin door) and canal system, Radiation Monitoring	<u>Radiological Protection Program, Event Response Procedures, Fuel Receipt & Shipping Program, Cask/Fuel Receiving Requirements</u>	Compromises the confinement capability of RBOF Facility, property damage	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	9

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
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Postulated event occurrence												
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level	
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety	Risk Rank
20	E-3	Earthquake, tornado or missile driven missiles damage RBOF building exhaust Filters/Filter room resulting in a release of radionuclides from damaged building exhaust filters to the environment	Natural Phenomena	None	None	U 4	Visual, audible, vibration	Shut down of the exhaust system results in automatic shutdown of the supply system	<u>Radiological Protection Program</u> (100 Ci of Cs-137 loading on filter), <u>Event Response Procedures</u>	Compromises the confinement capability of RBOF Facility, cask venting and decontamination activities secured, property damage	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	11
21	E-3	Release of radionuclides from RBOF building exhaust Filter through the stack	Filter failure, roughing filter failure, improper maintenance, inadequate procedures, worker error, stack collapse, loss of containment	Roughing filters installed to extend serviceability of building exhaust filters.	DOP testing of building exhaust filters, use of activated charcoal absorbers during fuel cutting, preventive maintenance	A 3	DOP test, visual,	Replaceable building exhaust filters	<u>Radiological Protection Program</u> (100 Ci of Cs-137 loading on filter), <u>Event Response Procedures</u>	Cask venting and decontamination activities secured	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	10
22	E-4	Fuel raised above basin water level	Equipment failure, inadequate procedure, worker error	Cranes require manual controls to raise fuel, radiation monitors.	NOPs, 8Q Safety Program, NSDS TSRs, NSDS tool depth markings	A 3	Visual and Audible Radiation Monitoring	Remote operations, shield walls, use bolt s spaced to maintain 4.5 feet of water above fuel, <u>NIMs</u>	<u>Radiological Protection Program</u> , <u>Event Response Procedures</u> , <u>Fuel Receipt & Shipping Program</u> , <u>Cask/Fuel Receiving Requirements</u>	Increase in radiation results in affected areas of operation being secured	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	7

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Postulated event occurrence			Risk Rank
				Design	Administrative			Design	Administrative	Consequence level	
23	E-4	Damaged cask received by rail or truck exposing workers to radioactive contamination	Breach of containment, material failure, over pressurization, thermal stress, mechanical damage	Cask design	Fuel Receipt & Shipping Program (Receipt of casks with known damage requires approval before shipment to RBOF, DOT and NRC regulations)	A 1	HP surveys, visual inspection	RBOF Confinement capability, <u>Cask Design</u>	<u>Radiological Protection Program</u> , <u>Fuel Receipt & Shipping Program</u>	Contamination of transport vehicle, handling equipment, Cask port Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	7
24	E-5	Dry cask goes critical while being immersed or filled with water	Modifications to a shipping cask, fuel array not critically safe	<u>Cask and fuel load geometry specifications</u> , cask shielding	Filling and immersion of cask as per procedures. SARP/COFC for the cask; <u>Fuel Receipt & Shipping Program</u>	BEU 2	Visual and Audible Radiation Monitoring	RBOF confinement capability, cask shielding, cask design, Control Room Concrete Wall, NIMs	<u>Radiological Protection Program</u> , <u>Event Response Procedures</u>	RBOF Facility is shutdown and evacuated, Other Facilities in the 200H Area may be shutdown and evacuated Moderate consequence. > 25 rem or life threatening injury to the onsite worker.	13
25	E-4	Fuel received not as described on shipping manifest exposing workers to unspecified penetrating radiation	Procedure or DOT violation by shipper	None	NOPs, TSRs, Receiving Requirements, Fuel Receipt & Shipping Program (paperwork sent before shipment)	A 1	Review of definitive loading list and shipment description	Reinforced Concrete Shield Walls, RBOF Confinement capability	<u>Radiological Protection Program</u> , <u>Event Response Procedures</u>	Additional fuel can not be received or other fuel shipped Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	7

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
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Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Postulated event occurrence			Risk Rank
				Design	Administrative			Design	Administrative	Consequence level	
26	E-4	Worker received crushing injury from object dropped from crane	Chain fall breaks or other equipment failure, worker error, inadequate procedures, natural phenomena, extrinsic event	Crane and related lifting devices design, Industrial Safety Standards	Hoisting and Rigging Program, NOPs, 8Q Safety Program	A I	Visual, audible	None	Onsite medical treatment, emergency team response	Possible property damage Moderate consequence. Standard Industrial Hazard.	4
26a	E-4	Worker received crushing injury from tractor trailer or train while delivering fuel	Equipment failure, inadequate procedures, worker error	Train tracks, RBOF and door construction and location, SRS security fences and gates	NOPs, 8Q Safety Program	A I	Visual, audible	None	Onsite maintenance worker, spare parts inventory	Compromises the confinement capability of RBOF Facility, property damage, downtime Moderate consequence. Standard Industrial Hazard.	4
27	E-4	Worker receives crushing injury from crane load in lateral motion	Worker error, equipment failure	Crane and related lifting devices design, Industrial Safety Standards	Hoisting and Rigging Program, NOPs, 8Q Safety Program	U I	Visual, audible	None	Onsite medical treatment, emergency team response	Possible property damage Moderate consequence. Standard Industrial Hazard.	5
28	E-4	First aid injuries from hand tool use	Worker error, tool failure, using non-approved tools or equipment, failure to use safety glasses, procedure violation	Industrial Safety Standards	Preventive maintenance, NOPs, 8Q Safety Program	A I	Visual, audible, verbal	None	Onsite medical treatment, emergency team response	Possible property damage Low consequence. Standard Industrial Hazard.	7

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
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Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
29	E-4	Steam burn during cask cooling	Steam line break, worker error	Engineering Standards, Industrial Safety Standards	Preventive maintenance, NOPs, 8Q Safety Program	A 1, 2	Visual, audible	None	Onsite medical treatment, emergency team response	None	Low consequence. Standard Industrial Hazard.	7	
30	E-4	Electrical shock	Worker error, improper lock and tagout, wire insulation failure, maintenance activities, contact with electrical circuits	Engineering Standards, Industrial Safety Standards	8Q Safety Program, preventive maintenance, NOPs	A 1, 2	Visual, audible, smell	Electrical codes	Onsite medical treatment, emergency team response	Possible property damage	Moderate consequence. Standard Industrial Hazard.	4	
31	E-4	Other first aid injuries	Falls or slips from splashed water, pinches, failure to wear safety glasses	Industrial Safety Standards, tool designs	8Q Safety Program	A 1, 2	Visual, verbal reports	Crane design, building design	Onsite medical treatment, emergency team response	Possible property damage	Low consequence. Standard Industrial Hazard.	7	
32	E-4	Worker falls in basin filled with water	Tripping, falling, struck by moving equipment, maintenance on railings around basins	Railings, gratings, partitions and plastic sheeting around basins or over canals	Two men on bridge rule, 8Q Safety Program (rails)	A 3	Visual, audible, verbal reports	Life throw rings, established walkways	Onsite medical treatment, emergency team response, <u>Basin Water Chemistry Control Program</u>	None	Moderate consequence. Standard Industrial Hazard. Negligible radiological consequence.	10	
33	E-4	Burn during welding	Worker error, equipment failure	Design Standards, Industrial Safety Standards	8Q Safety Program, Fire Protection Program (welding safety)	A 1, 2	Visual, audible, verbal	None	Onsite medical treatment, emergency team response	None	Low consequence. Standard Industrial Hazard.	7	

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
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Postulated event occurrence												
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Risk Rank	
				Design	Administrative			Design	Administrative	Consequence level		
33a	E-4	Earthquake damages RBOF structurally causing debris to crush worker	Natural Phenomena	Design Standards	None	U 4	Visual, audible, vibration	RBOF structure, shield walls, basins below grade	<u>Event Response Procedures</u>	Compromises the confinement capability of RBOF Facility, property damage	Moderate consequence. Standard Industrial Hazard.	5
33b	E-4	Tornado or tornado driven missiles damages RBOF structurally causing debris to crush worker	Natural Phenomena	Design Standards	None	EU 4	Visual, audible	RBOF structure, shield walls, basins below grade	<u>Event Response Procedures</u>	Compromises the confinement capability of RBOF Facility, property damage	Moderate consequence. Standard Industrial Hazard.	6
34	E-5	Floating fuel container resulting in criticality	Storage violation, equipment malfunction, defective container	Bulk density of canister, Rack, Bundle, Bucket, Inserts, Blocking Devices, and Hardware Design.	<u>Fuel Receipt & Shipping Program</u> , <u>Cask/Fuel Receiving Requirements</u> , <u>Nuclear Criticality Safety Program</u> (canning procedures), <u>Criticality Control Requirements</u> (store fuel property), <u>NSDS</u>	U 2	Visual and Audible Radiation Monitoring	Basin water shielding and decontamination factor, Basin and Canal System, walls shielding, NIMs, Control Room Concrete Wall	<u>Fuel Receipt & Shipping Program (RFA)</u> , <u>Cask/Fuel Receiving Requirements</u> , <u>Radiological Protection Program</u> , <u>Event Response Procedures</u>	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence. > 25 rem or life threatening injury to the onsite worker.	5

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Postulated event occurrence			Risk Rank
				Design	Administrative			Design	Administrative	Consequence level	
35	E-5	Fuel storage infraction results in criticality	Procedure violation or worker error	<u>Rack, Bundle, Bucket, Insert, Blocking Devices, and Hardware Design.</u>	<u>Nuclear Criticality Safety Program, Criticality Control Requirements (store fuel properly), NSDS</u>	U 2	Visual and Audible Radiation Monitoring	<u>Basin water shielding and decontamination factor, Basin and Canal System, walls shielding, NIMs</u>	<u>Fuel Receipt & Shipping Program (RFA), Cask/Fuel Receiving Requirements, Radiological Protection Program, Event Response Procedures</u>	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown Moderate consequence, > 25 rem or life threatening injury to the onsite worker.	5
36	E-5	Deformation of Fuel Storage Rack results in criticality	Roof and/or 100-ton crane falls into Storage Basin due to external load dropped from construction crane	<u>RBOF Building Design, Rack, Bundle, Bucket, Insert, Blocking Devices, and Hardware Design, Crane Limit Switch, Mechanical Stop.</u>	<u>Criticality Control Requirements (store fuel properly), Nuclear Criticality Safety Program, NSDS, Hoisting and Riggering Program</u>	EU 2	Visual and Audible Radiation Monitoring	<u>Basin water shielding and decontamination factor, Basin and Canal System, walls shielding, NIMs</u>	<u>Fuel Receipt & Shipping Program (RFA), Cask/Fuel Receiving Requirements, Radiological Protection Program, Event Response Procedures</u>	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown Moderate consequence, > 25 rem or life threatening injury to the onsite worker.	6

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
37a	E-5	Deformation of Fuel Storage Rack results in criticality	Mechanical failure causing crane or crane load to fall into basin.	<u>Crane Limit Switch</u> , <u>Mechanical Stop</u> , <u>Rack, Bundle, Bucket Inserts</u> , <u>Blocking Devices</u> , and <u>Hardware Design</u> .	<u>Criticality Control Requirements</u> (store fuel properly), <u>Nuclear Criticality Safety Program</u> , <u>NSDS</u> , <u>Hoisting and Rigging Program</u>	EU 2	Visual and Audible Radiation Monitoring	<u>Basin water shielding and decontamination factor</u> , <u>Basin and Canal System</u> , <u>walls shielding</u> , <u>NIMs</u>	<u>Fuel Receipt & Shipping Program (RFA)</u> , <u>Cask/Fuel Receiving Requirements</u> , <u>Radiological Protection Program</u> , <u>Event Response Procedures</u>	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence. > 25 rem or life threatening injury to the onsite worker.	6	
37b	E-5	Deformation of Fuel Storage Rack results in criticality	Worker error causes hoist hook tool to snag rack, dropped fuel	<u>Rack, Bundle, Bucket Inserts</u> , <u>Blocking Devices</u> , and <u>Hardware Design</u> .	<u>Nuclear Criticality Safety Program</u> (prevent rack snagging), <u>Criticality Control Requirements</u> (store fuel properly), <u>NSDS</u>	U 2	Visual and Audible Radiation Monitoring	<u>Basin water shielding and decontamination factor</u> , <u>Basin and Canal System</u> , <u>walls shielding</u> , <u>NIMs</u>	<u>Fuel Receipt & Shipping Program (RFA)</u> , <u>Radiological Protection Program</u> , <u>Event Response Procedures</u> , <u>Cask/Fuel Receiving Requirements</u>	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence. > 25 rem or life threatening injury to the onsite worker.	5	
38	E-5	Critical configuration in fuel bundle cause criticality	Worker error in assembling fuel bundle	<u>Bundle Design</u> , <u>Buckets and Bucket Inserts</u>	<u>Criticality Control Requirements</u> (prevent bundling infraction), <u>NSDS</u> , <u>Nuclear Criticality Safety Program</u>	U 2	Visual and Audible Radiation Monitoring	<u>Basin water shielding and decontamination factor</u> , <u>Basin and Canal System</u> , <u>walls shielding</u> , <u>NIMs</u>	<u>Fuel Receipt & Shipping Program (RFA)</u> , <u>Cask/Fuel Receiving Requirements</u> , <u>Radiological Protection Program</u> , <u>Event Response Procedures</u>	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence. > 25 rem or life threatening injury to the onsite worker.	5	

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Postulated event occurrence			Risk Rank	
				Design	Administrative			Design	Administrative	Impact on other systems		Consequence level
39	E-5	Configuration of fissile debris on basin floor causes criticality	Fuel ruptured, damaged, or corroded	<u>Rack, Bundle, Bucket, Bucket Inserts, Blocking Devices, and Hardware Design.</u>	<u>Nuclear Criticality Safety Program</u> (no disassembly without NCSE), <u>Criticality Control Requirements</u> (store fuel properly), <u>Basin Water Chemistry Control Program</u> (prevent corrosion), <u>NSDS</u>	U 2	Visual and Audible Radiation Monitoring	<u>Basin water shielding and decontamination factor, Basin and Canal System, walls shielding, NIMs</u>	<u>Fuel Receipt & Shipping Program (RFA), Cask/Fuel Receiving Requirements, Radiological Protection Program, Event Response Procedures</u>	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence. > 25 rem or life threatening injury to the onsite worker.	5

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence												
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level	Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems		
40	E-5	Fuel spills from fuel basket during fuel movement causing criticality	Basket dropped or ruptured due, basket lid off during moving, worker error, equipment malfunction	Cask and fuel load geometry specifications, Bucket and Design	Fuel Receipt & Shipping Program, Nuclear Criticality Safety Program (prevent fuel spills), Criticality Control Requirements, NSDS, NOPs (for casks being unloaded), Hoisting and Rigging Program	U 2	Visual and Audible Radiation Monitoring	Basin water shielding and decontamination in factor, Basin and Canal System, walls shielding, NIMs	Fuel Receipt & Shipping Program (RFA), Cask/Fuel Receiving Requirements, Radiological Protection Program, Event Response Procedures	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence, > 25 rem or life threatening injury to the onsite worker.	5
40a	E-5	Fuel spills from shipping cask during fuel loading/unloading operations resulting in criticality on cask basin floor	Equipment malfunction or operator error causes cask lid to fall onto cask. Other equipment failure or operator error causes cask contents to spill	Cask and fuel load geometry specifications.	Fuel Receipt & Shipping Program, Criticality Control Requirements (store fuel properly), Nuclear Criticality Safety Program, NSDS, Hoisting and Rigging Program	U 2	Visual and Audible Radiation Monitoring	Basin water shielding and decontamination in factor, Basin and Canal System, walls shielding, NIMs	Fuel Receipt & Shipping Program (RFA), Cask/Fuel Receiving Requirements, Radiological Protection Program, Event Response Procedures	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence, > 25 rem or life threatening injury to the onsite worker.	5

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
40b	E-5	Fuel handling/counting infraction in RRFC results in a criticality on basin floor	Procedure violation or operator error	<u>RRFC Safety Skirt, Rack, Bundles, Buckets, Bucket Inserts, Blocking Devices, and Hardware Design.</u>	<u>Criticality Control Requirements</u> (handle safe number, store fuel properly), <u>Nuclear Criticality Safety Program, NSDS, Fuel Receipt & Shipping Program, Cask/Fuel Receiving Requirements</u>	BEU 2	Visual and Audible Radiation Monitoring	Basin water shielding and decontamination factor, Basin and Canal System, walls shielding, NIMs	Fuel Receipt & Shipping Program (RFA), Cask/Fuel Receiving Requirements, Radiological Protection Program, Event Response Procedures	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence. > 25 rem or life threatening injury to the onsite worker.	13	
41	E-5	More than one basket out of cask at same time resulting in criticality	Procedure violation	None	<u>Criticality Control Requirements</u> (handle safe number), <u>NSDS, Nuclear Criticality Safety Program</u>	U 2	Visual and Audible Radiation Monitoring	<u>Basin water shielding and decontamination factor, Basin and Canal System, walls shielding, NIMs</u>	Fuel Receipt & Shipping Program (RFA), Cask/Fuel Receiving Requirements, Radiological Protection Program, Event Response Procedures	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence. > 25 rem or life threatening injury to the onsite worker.	5	

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
42	E-5	Low water level in basin allows decay heat to deform fuel to unsafe geometry causing criticality	Failure to add water to basin due to leak in water supply line or failure of level controller	<u>Rack, Bundle, Bucket, Insert, Blocking Devices, and Hardware Design,</u>	<u>Fuel Receipt & Shipping Program</u> (maximum heat load), <u>Cask/Fuel Receiving Requirements, Criticality Control Requirements</u> (store fuel properly), <u>Nuclear Criticality Safety Program, NSDS</u>	BEU 2	Visual and Audible Radiation Monitoring	Basin water shielding and decontamination factor, Basin and Canal System, walls shielding, NIMs	Fuel Receipt & Shipping Program (RFA), Cask/Fuel Receiving Requirements, Radiological Protection Program, Event Response Procedures	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	High consequence, ≥ 0.5 rem at site boundary.	13	
42a	E-5	Fast leak in 70 ton shipping cask adds positive reactivity for some fuel types resulting in criticality	Natural phenomena, vehicular impact, operator or maintenance error, mishandling, equipment failure	<u>Design of 70 ton shipping cask</u>	<u>Shipping cask procedures, Fuel Receipt & Shipping Program, Nuclear Criticality Safety Program, NSDS</u>	U 2	Visual and Audible Radiation Monitoring	<u>Control Room Wall Shielding, NIMs</u>	<u>Radiological Protection Program, Event Response Procedures</u>	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence, > 25 rem or life threatening injury to the onsite worker.	5	

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
43	E-5	Low water level in basin allows decay heat to deform fuel to unsafe geometry causing criticality	Leakage of water from basin to subsoil due to earthquake or rack failure	<u>Basin Design, Rack, Bundle, Bucket, Insert, Blocking Devices, and Hardware Design.</u>	<u>Fuel Receipt & Shipping Program</u> (maximum heat load), <u>Cask/Fuel Receiving Requirements, Criticality Control Requirements</u> (store fuel properly), <u>Nuclear Criticality Safety Program, NSDS</u>	EU 2	Visual and Audible Radiation Monitoring	Basin water shielding and decontamination factor, Basin and Canal System, walls shielding, NIMs	<u>Fuel Receipt & Shipping Program (RFA), Cask/Fuel Receiving Requirements, Radiological Protection Program, Event Response Procedures</u>	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	High consequence. ≥ 0.5 rem at site boundary.	3	
44	E-5	Disassembled pieces are arranged in an unsafe geometry on basin floor resulting in a criticality	Worker error during disassembly, procedure violation	<u>Rack, Bundles, Buckets, Bucket Inserts, Blocking Devices, and Hardware Design.</u>	<u>Criticality Control Requirements</u> (fuel stored properly), <u>NSDS, Nuclear Criticality Safety Program</u> (no disassembly without NCSE)	U 2	Visual and Audible Radiation Monitoring	<u>Basin water shielding and decontamination factor, Basin and Canal System, walls shielding, NIMs</u>	<u>Fuel Receipt & Shipping Program (RFA), Cask/Fuel Receiving Requirements, Radiological Protection Program, Event Response Procedures</u>	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence. > 25 rem or life threatening injury to the onsite worker.	5	
45	E-6	Loss of steam to RBOF	Leak, power outage, problem at source, outages, repairs	Engineering Standards	Preventive maintenance, routine testing	A 1	Monitors, visual, audible	None	Normal shutdown of all affected operations	Delay in cooling or decontaminating cask	Negligible consequence. Standard Industrial Hazard	10	

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
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Postulated event occurrence												
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level	Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems		
46	E-6	Loss of power to RBOF	Power failure, outages, vehicle repairs, vehicle crash, natural phenomena	Emergency generators for essential alarms and operations	Preventive maintenance	A I	Visual, monitors, verbal	Battery powered emergency lights	Normal shutdown of all affected operations	Unable to maintain proper ventilation, downtime	Negligible consequence. Standard Industrial Hazard.	10
47	E-6	Loss of purified water to RBOF	Leak, transfer error, spill, pump failure, plugged filter or tubes, repairs, problem at power house, natural phenomena	Holding tank on roof maintains a limited resource	Preventive maintenance, limits on water level, routine testing	A I	Basin level control system and monitors	None	Emergency safety procedures, normal shutdown of affected operations	Downtime	Negligible consequence. Standard Industrial Hazard.	10
48	E-6	Loss of Breathing Air	Leak, outages, repairs, compressor failure, loss of cooling water to compressor	Backup system	Preventive maintenance, routine testing	A I	Audible	None	Emergency safety procedures, normal shutdown of affected operations	Secure operations requiring use of breathing air	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	10
49	E-6	Loss of Instrument Air	Leak, outages, repairs, compressor failure, loss of cooling water to compressor	Backup system	Preventive maintenance, routine testing	A I	Audible	None	Emergency safety procedures, normal shutdown of affected operations	Downtime, minor property damage	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	10

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
51	E-6	Violent reaction between offsite fuels containing any material that reacts violently with water (e.g., sodium and potassium)	Aluminum storage container leaks	Sodium fuels sealed in single wall containers and welded	<u>Fuel Receipt & Shipping Program</u> (no fuels with any material that reacts violently with water (e.g., sodium and potassium), <u>Cask/Fuel Receiving Requirements</u>)	A I	Visual and Audible Radiation Monitoring	Confinement properties of RBOF, containment of containers, Radiation Monitoring	<u>Fuel Receipt & Shipping Program</u> (RFA inventory), <u>Radiological Protection Program</u> , <u>Event Response Procedures</u>	Shutdown of affected area operations, water damage to other equipment in the area, property damage	Moderate consequence. > 25 rem or life threatening injury to the onsite worker.	4	
52	E-8	NIM alarm malfunction	Electromagnetic interference, bypass during maintenance activities	Backup units, individual battery packs, alarms located in pairs	Bypass NIM, NIM surveillances, fissile material operations are immediately shutdown	A I	Audible, verbal report	Confinement properties of RBOF	<u>Radiological Protection Program</u> (no fuel movement without radiation monitoring)	Automatic shutdown of RBOF exhaust and supply systems	No consequence	10	
53	E-6	Failure of Radiation Monitor	Monitor malfunction	Backup monitors	Emergency shutdown of operations and building, routine testing, preventive maintenance	A I	Audible, visual	Confinement properties of RBOF	<u>Radiological Protection Program</u> (no fuel movement without radiation monitoring (fixed or portable))	None	No consequence	10	
54	E-6	Rail unable to ship fuel out of RBOF	Rail car malfunction, broken or damaged track	Stored material is kept in critically safe configuration in basins	Preventive maintenance	A I	Visual	Confinement capabilities of cask, RBOF and basin water	NOPs	Delay in schedule	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	10	

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence												
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level	Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems		
55	E-6	Truck unable to ship fuel out of RBOF	Mechanical failure of truck	Stored material is kept in critically safe configuration in basins	Preventive maintenance	A 1	Observation	Backup trucks/trailers	Onsite maintenance capabilities, spare parts	Delay in schedule	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	10
56	E-6	Fire spreads from nearby building	Nearby facility burning	Transit walls, distance to nearest facility	<u>Fire Protection Program</u> (limits on combustibles), <u>Event Response Procedures</u>	BEU 3	Area fire alarms, public address system	None	Event Response Procedures, Fire Protection Program	Shutdown of affected area operations, water damage to other equipment in the area, property damage	Low consequence. Standard Industrial Hazard.	13
57	E-7	Inability to transfer liquid waste from cask wash pit to waste tank in waste cell	Waste tank is filled to capacity, pluggage, pump failure	Cell and Transfer lines Design	NOPs	A 3	Visual, level monitors	RBOF and basin containment and confinement features	NOPs, waste tank trailers, <u>Radiological Protection Program</u>	Secure activities in Cash Wash Pit	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	10
58	E-7	Improper disposal of contaminated solid waste (equipment, cropped fuel assembly pieces)	Failure to determine radiation intensity prior to packaging, placing waste in clean dumpster, procedure violation, worker error	Identification of radiation controlled areas in building, barricades, labeling	Waste Management Program (directs what can be sent, WAC, and requires signatures of the inventory packages), <u>Radiological Protection Program</u>	A 1	HP monitoring	SRS site and area boundaries	NOPs, transport and receiving procedures for solid waste, <u>Radiological Protection Program</u>	None to RBOF's structure or equipment	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker. (No consequences to RBOF/RRF receptors.)	10

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
59	E-8	Worker contamination during maintenance	Leaks, spills, maintenance activities outside of ALARA philosophy	Shielding walls	NOPs, Radiological Protection Program (ALARA reviews)	A I	Alarms, verbal reports, HP surveys	RBOF and basin water shielding capability	<u>Radiological Protection Program</u> , onsite medical treatment	None	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	7	
59a	E-8	Exposure to penetrating radiation during maintenance	Leaks, spills, maintenance activities outside of ALARA philosophy	Shielding walls	NOPs, Radiological Protection Program (ALARA reviews)	A I	Alarms, HP surveys	RBOF and basin water shielding capability	<u>Radiological Protection Program</u> , onsite medical treatment	None	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	7	
60	E-8	Fire - ignition of plastic during welding	Welding sparks	None	NOPs, Fire Protection Program (welding safety and fire watcher)	A I	Visual, smell	Facility sections separated by partitions and shield walls, fire extinguishers, manual fire fighting capability, emergency exits	<u>Fire Protection Program</u> , <u>Event Response Procedures</u>	Shutdown of affected area operations, water damage to other equipment in the area, property damage	Low consequence. Standard Industrial Hazard.	7	

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
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Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
61	E-8	Electrical shock	Performing maintenance which requires exposure to live wire, procedure violation, worker error, improper lock and tagout, wire insulation failure	Engineering Standards, Industrial Safety Standards	8Q Safety Program, preventive maintenance, NOPs	A 1, 2	Visual, audible, smell	Electrical codes	Onsite medical treatment, emergency team response	Possible property damage	Moderate consequence. Standard Industrial Hazard.	4	
62	E-9	Aircraft crashes into roof causing roof collapse and deforms safe fuel configuration into an unsafe geometry resulting in a criticality/fire.	Extrinsic event	Facility Design (relatively small footprint)	Restricted Air Space	BEU 2	Visual and Audible Radiation Monitoring	Shielding capability of RBOF structure and basin water, rack configuration, RBOF's ventilation and exhaust system, Radiation Monitoring	Event Response Procedures, Fire Protection Program, Radiological Protection Program	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence. > 25 rem or life threatening injury to the onsite worker.	13	

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
63	E-10	Earthquake, tornado, high wind, causes damage to the basin walls, or storage racks, roof collapse, or crane derails resulting in stored fuel collecting in an unsafe geometry resulting in a criticality	Natural Phenomena	<u>Basin Design</u> , <u>Rack, Bundle</u> , <u>Bucket, Bucket Inserts</u> , <u>Blocking Devices</u> , and <u>Hardware</u> <u>Design</u> , <u>Crane Limit Switch</u> , <u>Mechanical Stop</u> , <u>Design of RBOF Building</u> (fragility curve)	<u>Hoisting and Rigging Program</u> , <u>Criticality Control Requirements</u> (store fuel properly), <u>Nuclear Criticality Safety Program</u> , <u>NSDS</u>	U 4	Noticeable ground movement, Radiation Monitoring	<u>Basin Water Shielding and DE</u> , <u>Basin and Canal Systems</u> , <u>Walls Shielding</u> , <u>NIMs</u>	<u>Cask/Fuel Receiving Requirements</u> , <u>Event Response Procedures</u> , <u>Fuel Receipt & Shipping Program</u> (RFA inventory)	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence. > 25 rem or life threatening injury to the onsite worker.	5	
63a	E-10	Earthquake, tornado, high wind, causes damage to the basin walls, or storage racks, roof collapse, or crane derails resulting in stored fuel collecting in an unsafe geometry resulting in a criticality and basin in draindown condition	Natural Phenomena	<u>Design of RBOF Building</u> (fragility curve), <u>Basin Design</u> , <u>Rack, Bundle</u> , <u>Bucket, Bucket Inserts</u> , <u>Blocking Devices</u> , and <u>Hardware</u> <u>Design</u> , <u>Crane Limit Switch</u> , <u>Mechanical Stop</u>	<u>Hoisting and Rigging Program</u> , <u>Criticality Control Requirements</u> (store fuel properly), <u>Nuclear Criticality Safety Program</u> , <u>NSDS</u>	EU 4	Noticeable ground movement, Radiation Monitoring	<u>Basin Water Shielding and DE</u> , <u>Basin and Canal Systems</u> , <u>Walls Shielding</u> , <u>NIMs</u>	<u>Event Response Procedures</u> , <u>Fuel Receipt & Shipping Program</u> (RFA inventory), <u>Cask/Fuel Receiving Requirements</u>	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	High consequence. ≥ 0.5 rem at site boundary.	3	

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
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Postulated event occurrence												
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features		Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety	
64	E-10	Tornado or high wind causes criticality to occur when portions of the roof collapses into the storage basins resulting in stored fuel collecting in an unsafe geometry	Natural Phenomena	<u>Design of RBOF Building</u> (fragility curve), <u>Rack, Bundle, Bucket</u> , <u>Inserts</u> , <u>Blocking Devices</u> , and <u>Hardware</u> <u>Design, Crane Limit Switch, Mechanical Stop</u>	<u>Criticality Control Requirements</u> (store fuel properly), <u>Nuclear Safety Program</u> , <u>NSDS</u> , <u>Hoisting and Rigging Program</u>	U 4	Visual and Audible Radiation Monitoring	<u>Basin water shielding and decontamination factor, Basin and Canal System</u> , walls shielding, NIMs	<u>Fuel Receipt & Shipping Program</u> (RFA inventory), <u>Event Response Procedures</u> , <u>Radiological Protection Program</u> , <u>Cask/Fuel Receiving Requirements</u>	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence. >25 rem or life threatening injury to the onsite worker.	5
65	E-10	Tornado or high winds impels missile into storage basins causing a criticality by dislodging stored fuel into an unsafe geometry	Natural Phenomena	<u>Basin Water</u> (prevents missile damage), <u>Rack, Bundle, Bucket</u> , <u>Inserts</u> , <u>Blocking Devices</u> , and <u>Hardware</u> <u>Design, Crane Limit Switch, Mechanical Stop</u>	<u>Cask/Fuel Receiving Requirements</u> , <u>Criticality Control Requirements</u> (store fuel properly), <u>Nuclear Safety Program</u> , <u>NSDS</u> , <u>Hoisting and Rigging Program</u>	EU 4	Visual and Audible Radiation Monitoring	<u>Basin water shielding and decontamination factor, Basin and Canal System</u> , walls shielding, NIMs	<u>Fuel Receipt & Shipping Program</u> (RFA), <u>Event Response Procedures</u> , <u>Radiological Protection Program</u> , <u>Cask/Fuel Receiving Requirements</u>	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence. >25 rem or life threatening injury to the onsite worker.	6

Table 3.3-13 Hazard Evaluation - RBOF Basin Area (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence												
		Prevention features				Freq. Level & Source	Method of detection	Mitigation features		Consequence level		Risk Rank
Event No.	Event Category	Causes	Design	Administrative	Design			Administrative	Impact on other systems	Health and Safety		
66	E-10	Heavy rains floods basins through roof leak resulting in flooding of basin floor	Natural Phenomena	<u>Building Design</u>	None	EU 3	Audible, visual	Shielding capability of RBOF structure and basin water, rack configuration, RBOF's ventilation and exhaust system	<u>Basin Water Chemistry Control Program</u> (activity limit), <u>Event Response Procedures</u>	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Negligible consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	9
67	E-10	Facility is damaged by hail, or ice storms resulting in containment breach	Natural phenomena	<u>Building 244-H Roof Design</u>	None	BEU 2	Audible, visual	Building design, Engineering Standards	----	Property damage	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	13
68	E-10	Meteorite crashes into facility structure resulting in containment breach	Natural phenomena	<u>Building Design</u> (small footprint)	None	BEU 2	Audible, visual	Building design, Engineering Standards	----	Property damage	Negligible consequence.	13
69	E-10	Lightning strikes facility resulting in damage to structure and/or components	Natural phenomena	<u>Building Design</u>	None	A 2	Audible, visual	Building design, Engineering Standards	----	Property damage,	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	10

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence												
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level	
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety	Risk Rank
310	E-1	Resin fire resulting in a radioactive airborne release at ground level	Loss of containment (leak, equipment malfunction, operator error) during transfer from the portable deionizer to the hose rack and evaporation dries out resin and resin ignites	Design of portable deionizer, hose rack, and associated hoses and valves, concrete pad sloping to drain which drains to waste cell, water available for flushing of pad	Reactor Deionizer Resin Replacement/Regeneration Program (alpha inventory control for the portable deionizer), HP surveys	A 1	Visual	None	Reactor Deionizer Resin Replacement/ Regeneration Program (inventory control for the portable deionizer), Radiological Protection Program, Fire Protection Program, Event Response Procedures	Shutdown of affected area operations, water or chemical damage to other equipment in the area, Moderate consequence. > 25 rem or life threatening injury to the onsite worker.	4	
311	E-1	Ignition of combustibles from mishandling of flammables normally stored in fire cabinet resulting in a RRF Control Room Fire	Unknown ignition source, worker error, material error, inadequate procedure	None	Fire Protection Program (Restriction of ignition sources and combustibles materials),	A 3	Smell, visual (Fire detectors are not installed)	Facility separated by shielding walls, fire extinguishers, manual fire fighting capability, emergency exits	Fire Protection Program, Event Response Procedures	Shutdown of affected area operations, water or chemical damage to other equipment in the area, property damage Low consequence. Standard Industrial Hazard.	7	
312	E-1	Vehicle fire at Hose Racks or Chemical Pads	Electrical short, unknown ignition source	Standard vehicle design	8Q Safety Program, Fire Protection Program (limit combustibles)	A 3	Smell, visual	Located outside	Fire Protection Program, Event Response Procedures	Shutdown of affected area operations, water or chemical damage to other equipment in the area, property damage Low consequence. Standard Industrial Hazard.	7	

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
313	E-1	Nitric acid promotes fire	Unknown ignition source, electrical short, fire, vehicle crash, fire from adjoining area spreads to chemical pad	None	Fire Protection Program (limit amount of combustibles)	EU 2	Smell, visual	Outside location on chemical pads	Event Response Procedures, Spill response, Fire Protection Program, Hazardous Material Inventory Control Program (inventory control on nitric acid)	Shutdown of affected area operations, water or chemical damage to other equipment in the area, property damage	Moderate consequence. >ERPG-2 or life threatening conditions exist.	6	
314	E-1	Sodium Nitrite explosion occurs when heated to >537°C	Unknown ignition source, electrical short, vehicle crash, fire from adjoining area spreads to chemical pads	None	Fire Protection Program (limit amount of combustibles)	EU 2	Audible, visual, smell	Located on chemical pads	Event Response Procedures, Spill response, Fire Protection Program, Hazardous Material Inventory Control Program (inventory control on sodium nitrite)	Shutdown of affected area operations, water or chemical damage to other equipment in the area, property damage	Moderate consequence. >ERPG-2 or life threatening conditions exist.	6	

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
314a	E-2	Breach of vessels or related pipes, valves, etc. on Chemical Pad No. 2 permits loss of greater than 167 gallons of 30% sodium nitrite	Material failure, mechanical damage, dropping of vessel during moving, equipment malfunction, worker error, overfilling, over pressurization, corrosion , natural phenomena, surface vehicle, rain shelter collapses	Elevated chemical pad, curbing, drains to waste tank	Weekly and semiannually inspections, preventive maintenance, NOPs, labeling of contents and hazards	A 1	Visual, smell, liquid level indicator	<u>Chem. Pad No. 2 berm and drain system design</u>	Reporting of spills to Area Emergency Coordinator for containment and cleanup activities, protective clothing, temporary dams, emergency response teams on-site, onsite medical treatment, <u>Hazardous Material Inventory Control Program, Event Response Procedures, Chem. Pad No. 2 pumping procedures, Spill Response</u>	None to shutdown of affected operations, chemical damage to other equipment	Moderate consequence. > ERPG-2 or life threatening injury to the onsite worker.	4	

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
314b	E-2	Breach of vessels or related pipes, valves, etc. on Chemical Pad No. 2 permits loss of less than 167 gallons of 30% sodium nitrite	Material failure, mechanical damage, dropping of vessel during moving, equipment malfunction, worker error, overfilling, over pressurization, corrosion, rain shelter collapses	Elevated chemical pad, curbing, drains to waste tank	Weekly and semiannually inspections, preventive maintenance, NOPs, labeling of contents and hazards	A 1	Visual, smell, liquid level indicator	Chem. Pad No. 2 berm and drain system design	Reporting of spills to Area Emergency Coordinator for containment and cleanup activities, protective clothing, temporary dams, emergency response teams on-site, onsite medical treatment, Hazardous Material Inventory Control Program, Event Response Procedures, Chem. Pad No. 2 pumping procedures, Spill Response	None to shutdown of affected operations, chemical damage to other equipment	Low consequence. $\leq$ ERPG-2 or non-life threatening injury to the onsite worker.	7	
315	E-2	Breach of vessels or related pipes, valves, etc. on Chemical Pad No. 1 permits loss of less than 72 gallons of 50% sodium hydroxide	Material failure, mechanical damage, dropping of vessel during moving, equipment malfunction, worker error, overfilling, over pressurization, corrosion, rain shelter collapses	Elevated chemical pad, curbing, drains to waste tank	Weekly and semiannually inspections, preventive maintenance, NOPs, labeling of contents and hazards	A 1	Visual, smell, liquid level indicator	Curbing, drain to waste cell	Reporting of spills to Area Emergency Coordinator for containment and cleanup activities, protective clothing, temporary dams, emergency response teams on-site, onsite medical treatment, Hazardous Material Inventory Control Program, Event Response Procedures, Spill Response	None to shutdown of affected operations, chemical damage to other equipment	Low consequence. $<$ ERPG-2 or only a minor injury to the onsite worker.	7	

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Postulated event occurrence			Risk Rank
				Design	Administrative			Design	Administrative	Consequence level	
316	E-2	Breach of vessels or related pipes, valves, etc. on Chemical Pad No. 1 permits loss of greater than 72 gallons of 50% sodium hydroxide	Material failure, mechanical damage, dropping of vessel during moving, equipment malfunction, worker error, overfilling, over pressurization, corrosion, natural phenomena, surface vehicle, rain shelter collapses	Elevated chemical pad, curbing, drains to waste tank	Weekly and semiannually inspections, preventive maintenance, NOPs, labeling of contents and hazards	A 1	Visual, smell, liquid level indicator	Curbing, drain to waste cell	Reporting of spills to Area Emergency Coordinator for containment and cleanup activities, protective clothing, temporary dams, emergency response teams on-site, onsite medical treatment, Hazardous Material Inventory Control Program, Event Response Procedures, Spill Response	None to shutdown of affected operations, chemical damage to other equipment to onsite worker. Moderate consequence. > ERPG-2 or life threatening injury to onsite worker.	4

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Postulated event occurrence			Risk Rank	
				Design	Administrative			Mitigation features		Consequence level		
								Design	Administrative	Impact on other systems		Health and Safety
317	E-2	Breach of vessels or related pipes, valves, etc. on Chemical Pad No. 1 permits loss of less than 71 gallons of 50% nitric acid	Material failure, mechanical damage, dropping of vessel during moving, equipment malfunction, worker error, overfilling, over pressurization, corrosion, rain shelter collapses	Elevated chemical pad, curbing, drains to waste tank	Weekly and semiannually inspections, preventive maintenance, NOPs, labeling of contents and hazards	A I	Visual, smell, liquid level indicator	Curbing, drain to waste cell	Reporting of spills to Area Emergency Coordinator for containment and cleanup activities, protective clothing, temporary dams, emergency response teams on-site, onsite medical treatment, <u>Hazardous Material Inventory Control Program, Event Response Procedures</u>	None to shutdown of affected operations, chemical damage to other equipment	Low consequence. < ERPG-2 or only a minor injury to the onsite worker.	7

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Postulated event occurrence			Risk Rank
				Design	Administrative			Design	Administrative	Consequence level	
318	E-2	Breach of vessels or related pipes, valves, etc. on Chemical Pad No. 1 permits loss of greater than 71 gallons of 50% nitric acid	Material failure, mechanical damage, dropping of vessel during moving, equipment malfunction, worker error, overfilling, over pressurization, corrosion, natural phenomena, surface vehicle, rain shelter collapses	Elevated chemical pad, curbing, drains to waste tank	Weekly and semiannually inspections, preventive maintenance, NOPs, labeling of contents and hazards	A 1	Visual, smell, liquid level indicator	Curbing, drain to waste cell	Reporting of spills to Area Emergency Coordinator for containment and cleanup activities, protective clothing, temporary dams, emergency response teams on-site, onsite medical treatment, Hazardous Material Inventory Control Program, Event Response Procedures, Spill Response	None to shutdown of affected operations, chemical damage to other equipment to onsite worker.	4
319	E-6	Loss of instrument air to hose racks and nitric acid dumpster filter regulator	Bad connection, source to RRF out, leaking valve or hose, worker error, compressor failure	Engineering Standards	Preventive maintenance, backup systems, routine testing	A 1	Flow and pressure meters, monitors, visual, audible	None	Normal shutdown of affected operations as necessary	Delay in operations	10

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features		Postulated event occurrence		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety	
321	E-2	Breach of portable deionizer while at RRF releasing contaminated water to subsurface	Faulty deionizer, over pressurization, thermal stress, material failure, overfilling, operator error, natural phenomena	Deionizer design, drain to waste tank, curbing	Regulated area, chains around jobsite, NOPs, preventive maintenance	A I	HP surveys, visual	Curbing and drain to waste cell	Reactor Basin Water Chemistry Control Program, Reactor Deionizer Resin Replacement/Regeneration Program (control of activity), Event Response Procedures, Radiological Protection Program, spill response	None to shutdown of affected operations, possible damage to other equipment	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	10
324	E-2	Contaminated water released to immediate area through leaking portable deionizer	Material failure, over pressurization, thermal stress, corrosion, mechanical damage, faulty deionizer, operator error	Deionizer design, drain to waste tank, curbing	Regulated area chains around jobsite, NOPs, preventive maintenance, labeling of valves on deionizer	A I	Visual and Audible Radiation Monitoring	Curbing, drain to waste cell, Radiation Monitoring	Reactor Basin Water Chemistry Control Program, Reactor Deionizer Resin Replacement/Regeneration Program (control of activity), Event Response Procedures, Radiological Protection Program, spill response	None to shutdown of affected operations, chemical damage to other equipment	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	10

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
328c	E-2	Breach of vessels or related pipes, valves, etc. permits loss of <30750 gallons of Ethylene Glycol	Material failure, mechanical damage, dropping of vessel during moving, equipment malfunction, worker error, overfilling, over pressurization, corrosion, natural phenomena, extrinsic events	Elevated chemical pad, curbing, drains to waste tank	Weekly and semiannually inspections, preventive maintenance, NOPs, labeling of contents and hazards	A 1	Visual, smell, liquid level indicator	Curbing, drain to waste cell	<u>Hazardous Material Inventory Control Program, Event Response Procedures</u>	None to shutdown of affected operations, chemical damage to other equipment	Low consequence. < ERPG-2 or only a minor injury to the onsite worker.	7	
329c	E-3	Tornado driven missile penetrates RRF shielded wall or the roof and enters the resin regeneration cell	Natural phenomena	None	None	EU 1, 2	Visual, audible	Cell Design	<u>Deionizer Resin Replacement/Regeneration Program</u> (activity loading), <u>Radiological Protection Program, Event Response Procedures</u>	Shutdown of affected area operations	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	9	
330	E-4	Fall from 2 ft step ladder while reading sight glass	Over extension, worker error, falls or slips	Industrial Safety Standards	8Q Safety Program	A 1, 2	Visual, verbal reports	None	Onsite medical treatment, emergency team response	Possible property damage	Low consequence. Standard Industrial Hazard.	7	

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence												
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features		Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety	
331	E-4	First aid injuries from hand tool use	Failure to wear safety glasses or other protective equipment, tool failure, worker error, using non-approved tools or equipment, pinches	Industrial Safety Standards	8Q Safety Program	A 1, 2	Visual, verbal reports	None	Onsite medical treatment, emergency team response	Possible property damage	Low consequence. Standard Industrial Hazard.	7
332	E-4	Other first aid injuries from normal operations	Falls or slips from splashed liquids, pinches, failure to wear safety glasses, or other protective equipment, tool failure, worker error	Industrial Safety Standards, WSRC Safety program	8Q Safety Program	A 1, 2	Visual, verbal reports	None	Onsite medical treatment, emergency team response	Possible property damage	Low consequence. Standard Industrial Hazard.	7
333	E-4	Electrical shock	Worker error, improper lock and tagout, wire insulation failure, maintenance activities	DuPont Engineering Standards, Industrial Safety Standards	8Q Safety Program	A 1, 2	Visual, audible, smell	Electrical codes	Onsite medical treatment, emergency response team	Possible property damage	Moderate consequence. Standard Industrial Hazard.	4

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Event No.	Event Category	Postulated Event Description	Causes	Prevention Features		Freq. Level & Source	Method of detection	Postulated event occurrence			Risk Rank
				Design	Administrative			Design	Administrative	Consequence level	
334	E-4	Breach of deionizers or associated pipes, hoses, or valves causing exposure to ionizing radiation during resin transfer	Leaks, spills	DuPont Engineering Standards, Industrial Safety Standards, radiation monitors	NOPs	A 1, 2	Area Radiation Monitors	Radiation Monitoring	<u>Event Response Procedures</u> , <u>Radiological Protection Program</u> , <u>Spill Response</u> , <u>Reactor Deionizer Resin Replacement/Regeneration Program</u> (control on inventory activity)	Shutdown of affected operations Low consequence, ≤ 25 rem or non-life threatening injury to the onsite worker.	7
335	E-4	Skin/eye burns from contact with chemicals	Pressurized hose not solidly connected, leaks/spills, routine maintenance, worker error, material or mechanical error, failure, or damage, over pressurization, thermal stress, corrosion, chemical reaction	Elevated chemical pad, curbing, drains to waste tank	8Q Safety Program	A 1, 2	Visual, smell, liquid level indicators	Curbing, drain to waste cell	Reporting of spills to Area Emergency Coordinator for containment and cleanup activities, protective clothing, temporary dams, emergency response team on-site, onsite medical treatment	None to shutdown of affected operations, chemical damage to other equipment Low consequence, Standard Industrial Hazard.	7
336	E-5	Criticality	No causes identified	----	----	----	----	----	----	----	N/A

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
337	E-6	Loss of steam to Chemical Pads	Leak, power outage, problem at source, outages, repairs	DuPont Engineering Standards	Preventive maintenance, routine testing	A 1	Monitors, visual, audible	None	Normal shutdown of all affected operations	Delay operations, remote possibility of NaOH tank or lines freezing	Negligible consequence. Standard Industrial Hazard.	10	
338	E-6	Loss of power to RRF	Power failure, outages, repairs, vehicle crash, natural phenomena	Standby generators for essential alarms and operations	Preventive maintenance	A 1	Visual, monitors, verbal	Battery powered emergency lights	Normal shutdown of all affected operations	Unable to maintain proper ventilation, downtime	Negligible consequence. Standard Industrial Hazard.	10	
339	E-6	Loss of instrument air	Leak, outages, repairs, compressor failure, loss of cooling water to compressor	Backup system	Preventive maintenance, routine testing	A 1	Audible	None	Emergency safety procedures, normal shutdown of affected operations	Downtime, minor property damage	Negligible consequence. Standard Industrial Hazard.	10	
340	E-6	Fire or explosion occurring in the waste tank farm could spread to the RRF	Nearby facility fire	Distance to nearest facility, structure walls	Fire Protection Program, NOPs, Event Response Procedures	EU 3	None	Facility separated by shielding walls, fire extinguishers, emergency exits	Event Response Procedures, Fire Protection Program	Shutdown of affected area operations, water or chemical damage to other equipment in the area, property damage	Low consequence. Standard Industrial Hazard.	9	
341	E-7	Inability to transfer liquid waste from RRF area to waste tanks in waste cell	Waste tank is filled to capacity, pluggage, pump failure, drain plug not removed, operator error	Cell and Transfer lines Design	NOPs	U 1, 2	Visual, level monitors	Waste tank and waste cell containment and confinement capabilities	NOPs, Radiological Protection Program	Secure activities in RRF area	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	11	

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
342	E-7	Improper disposal of contaminated solid waste (resin, equipment, replaced parts.)	Failure to determine radiation intensity prior to packaging, improperly packaging waste, worker error	Identification of radiation controlled areas, barricades, labeling, constant HP monitoring	Waste Management Program (directs what can be sent, WAC, and requires signatures of the inventory packages), Radiological Protection Program	A I	Hp monitoring	SRS site and area boundaries	NOPs, transport and receiving procedures for solid waste, <u>Radiological Protection Program</u>	None to RRF structure of equipment	Negligible consequence, < 0.5 rem or only a minor injury to the onsite worker (No consequence to RBOF/RRF receptors.)	10	
343	E-8	Chemical exposure during maintenance (spill less than RQ, TPQ, or TQ)	Splashed chemicals, failure to wear safety glasses, or other protective equipment, tool failure, worker error	None	NOPs, 8Q Safety Program	A 1, 2	Visual, verbal reports	None	<u>Hazardous Material Inventory Control Program (inventory), Event Response Procedures (spill response, chemical safety)</u>	Possible property damage	Moderate consequence. Standard Industrial Hazard.	4	

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
345	E-9	Aircraft crashes into Truck, Hose Box 1 or 2, or Chemical Pads, or into the RRF structure	Extrinsic event	Facility Design (relatively small footprint)	None	BEU 2	Visual, audible, possible alarms, smell	Facility separated by shielding walls, fire extinguishers, manual fire fighting capability, emergency exits, curbing, drain to waste cell	Event Response Procedures, Hazardous Material Inventory Control Program	Shutdown of affected area operations, water or chemical damage to other equipment in the area, property damage	Moderate consequence. > 25 rem, > ERPG-2 or life threatening injury to onsite worker.	13	
345a	E-9	Ground Vehicle crashes into trailer/portable deionizer staged at Hose Box 1 for resin regeneration resulting in breach of containment	Mechanical failure, operator error	Facility Design (location makes it hard for vehicle to reach)	Administrative controls (low speed), access control to area, NOPs	U 1	Visual and Audible Radiation Monitoring	Radiation Monitoring	Event Response Procedures, Radiological Protection Program, Spill Response, Reactor Deionizer Resin Replacement/Regeneration Program	Shutdown of affected area operations	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	8	
345c	E-9	Ground Vehicle crashes into Chemical Pad No. 1 resulting in breach of containment	Mechanical failure, operator error	Facility Design (location makes it hard for vehicle to reach)	Administrative controls (low speed), access control to area, NOPs	U 1	Visual, audible	Tank Design (limits inventory)	Hazardous Material Inventory Control Program (inventory control), Event Response Procedures, Spill Response	Shutdown of affected area operations	Moderate consequence. > ERPG-2 or life threatening injury to onsite worker.	5	

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
345d	E-9	Ground Vehicle crashes into Chemical Pad No. 2 resulting in breach of containment	Mechanical failure, operator error	Facility Design (location makes it hard for vehicle to reach)	Administrative controls, (low speed), access control to area, NOPs	U 1	Visual, audible	Tank Design (limits inventory)	Hazardous Material Inventory Control Program (inventory control), Event Response Procedures, Spill Response	Shutdown of affected area operations	Moderate consequence. > ERPG-2 or life threatening injury to the onsite worker.	5	
346	E-10	Meteorite or missile crashes into RRF, Chemical Pad or hose rack resulting in a containment breach	Natural Phenomena	None	None	BEU 2	Visual, audible, possible alarms	Facility separated by shielding walls, fire extinguishers, emergency exits, curbing, drain to waste cell	Event Response Procedures, Hazardous Material Inventory Control Program	Shutdown of affected area operations, water or chemical damage to other equipment in the area, property damage	Moderate consequence. > ERPG-2 or life threatening injury to onsite worker.	13	
347	E-10	Earthquake, high wind, tornado, etc. damages Chemical Pad or rain shelter resulting in a containment breach	Natural Phenomena	Seismic and Wind Design of Chem Pads, Berm, rainshelter, and Tanks, Chem Pad No. 2 Berm	None	U 4	Visual, audible, possible alarms	Facility separated by shielding walls, fire extinguishers, emergency exits, curbing, drain to waste cell	Event Response Procedures, Hazardous Material Inventory Control Program, Chem. Pad No. 2 pumping procedures, Spill Response	Shutdown of affected area operations, water or chemical damage to other equipment in the area, property damage	Moderate consequence. > ERPG-2 or life threatening injury to the onsite worker.	5	

Table 3.3-14 Hazard Evaluation - RRF Control Room, Target Wash Station, Chemical Pads, and Hose Racks (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence												
		Prevention features			Mitigation features					Consequence level		
Event No.	Event Category	Postulated Event Description	Causes	Design	Administrative	Freq. Level & Source	Method of detection	Design	Administrative	Impact on other systems	Health and Safety	Risk Rank
347a	E-10	Earthquake, high wind, tornado, damages RRF, hose rack or rain shelter resulting in a containment breach	Natural Phenomena	Cell Design	None	U	Visual, audible, possible alarms	Facility separated by shielding walls, fire extinguishers, emergency exits, curbing, drain to waste cell	<u>Reactor/RBOF</u> <u>Deionizer Resin Replacement/Regeneration Program</u> , <u>Event Response Procedures</u> , <u>Radiological Protection Program</u>	Shutdown of affected area operations, water or chemical damage to other equipment in the area, property damage	Low consequence. $\leq$ ERPG-2 chemicals and $<$ low consequence threshold of radionuclides released.	8
348	E-10	The Chemical Pads or hose racks are damaged by hail, or ice storms resulting in a containment breach	Natural Phenomena	Design of Tanks, Rain cover design	None	U	Visual, audible, possible alarms	<u>Chem. Pad No. 2 berm and drain design</u> , <u>Seismic and Wind Design of Chem Pads</u> , <u>Berms</u> , <u>rainshelters</u> , <u>and Tanks</u>	<u>Hazardous Material Inventory Control Program</u> (inventory control), <u>Event Response Procedures</u> , <u>Spill Response</u>	Shutdown of affected area operations, water or chemical damage to other equipment in the area, property damage	Moderate consequence. $>$ ERPG-2 or life threatening injury to the onsite worker.	5
349	E-10	Lightning strikes RRF resulting in damage to structure and/or components	Natural Phenomena	Building Design	None	U	Audible, visual		Minor property damage, possibility of fire	None	Negligible consequence. $<$ 0.5 rem or only a minor injury to the onsite worker.	11

Table 3.3-15 Hazard Evaluation - Waste Cell, Decontamination Cell, and RRF Cell

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
400	E-1	Fire/explosion of flammable vapors in waste cell Tank 38	Ignition of explosive mixture of hydrogen, oxygen, and air in cask, organic antifreeze (i.e., methanol) is not sufficiently diluted	Waste tank design	<u>Fuel Receipt & Shipping Program</u> (controls antifreeze - i.e., ethylene glycol and methanol), <u>Hazardous Material Inventory Control Program</u> (controls antifreeze - i.e., ethylene glycol and methanol)	EU 3	Visual, observation audible	Cask and equipment design, ventilation system, RBOF's containment capability	<u>Event Response Procedures, Fire Protection Program, Radiological Protection Program</u>	Stoppage of operations in RRF, Cask Wash Pit and or Cask Unloading Basin, property damage	Moderate consequence due to Hazard associated with a fire or explosion. Low radiological consequence. ≤ 25 rem to the onsite worker.	6	
400a	E-1	Explosion resulting from an uncontrolled reaction occurring from addition of undiluted nitric acid to resin promoting rapid resin degradation and increased thermal instability of resin during resin regeneration in RRF Cell.	Operator error, loss of deionized water, inadequate procedure, procedure violation, storage tank filled with incorrect chemical	Acid metering pump, vessel venting	Trained Operators, operating limits, flow rates of nitric acid and water, deionized water level limits. NOPs, Limited source term due to periodic resin replacement.	A 1	Worker observation audible	None	<u>Reactor/RBOF Deionizer Resin Replacement/Regeneration Program</u> (inventory of activity), <u>Event Response Procedures, Radiological Protection Program</u>	Shutdown of affected area operations	Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	7	

Table 3.3-15 Hazard Evaluation - Waste Cell, Decontamination Cell, and RRF Cell (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Postulated event occurrence			Risk Rank
				Design	Administrative			Design	Administrative	Consequence level	
400b	E-1	Resin fire resulting in a radioactive airborne release at ground level	RBOF resin allowed to dry and high curie loading in resin generates sufficient heat to ignite resin	Decon or Waste Cell Design (not many spark sources)	HP surveys, Basin Water Chemistry Control Program (self ignition is controlled by controlling alpha's in basin water)	EU 2	Visual	None	Deionizer Resin Replacement/Regeneration Program (inventory control for RBOF resin), Radiological Protection Program, Fire Protection Program, Event Response Procedures	Shutdown of affected area operations, water or chemical damage to other equipment in the area, Moderate consequence. > 25 rem or life threatening injury to the onsite worker.	6
401	E-2	Breach of waste tank or associated piping releases contaminated waste to waste cell	Material failure, corrosion, mechanical damage, overflow natural phenomena	DuPont Engineering Standards, confinement capability of waste cell, agitator, water eductor	Remotely located valve controls, preventive maintenance, NOPs	A 1	Liquid level dip tube	Cell Design (confines waste)	Deionizer Resin Replacement/Regeneration Program (inventory on activity), Event Response Procedures, Radiological Protection Program, Spill Response	Shutdown of affected area operations Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	10

Table 3.3-15 Hazard Evaluation - Waste Cell, Decontamination Cell, and RRF Cell (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Postulated event occurrence			Risk Rank
				Design	Administrative			Design	Administrative	Consequence level	
401a	E-2	Earthquake damage or wind induced failure of rain shelter causes nitrous dioxide (NO_2) gas to be generated due to nitric acid and sodium nitrite mixing in Waste Tank EP38 due to tank ruptures on Chemical Pads No. 1 and No. 2.	Natural phenomena	Sodium nitrite and nitric acid tank designs, Chem Pad No. 2, Chem Pad No. 2 pump and piping design mod	Hazardous Material Inventory Control Program (sodium nitrite inventory limit), Chem. Pad No. 2 Pumping Procedures	BEU 1	Liquid level dip tube, brown fumes flowing from RBOF ventilation stack	None	Event Response Procedures	Shutdown of affected area operations Moderate consequence. > ERPG-2 or life threatening injury to onsite worker.	13
402	E-2	Loss of containment in low level waste trailer or drums used to confine waste from cask wash pit	Natural phenomena, operator error, vehicle impact, corrosion	Engineering standards used to construct waste container	Preventive maintenance procedures and inspections, NOPs	A 2	Liquid level indication, visual	None	Waste Management Program (waste acceptance criteria set by receiving area), Event Response Procedures, Radiological Protection Program, Spill Response	Shutdown of affected area operations Low consequence. ≤ 25 rem or non-life threatening injury to the onsite worker.	7
402a	E-3	Loss of Confinement	No additional scenarios identified	----	----	----	----	----	----	----	N/A
403	E-4	Injuries to workers	No additional scenarios identified	----	----	----	----	----	----	----	N/A

Table 3.3-15 Hazard Evaluation - Waste Cell, Decontamination Cell, and RRF Cell (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence												
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level	
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety	Risk Rank
404	E-5	Fissile material on Basin Purification Filter or Deionizers, forming an unsafe geometry and causing a criticality	Equipment or material failure, inadequate procedures, worker error, thermal stress, corrosion, chemical reaction, mechanical damage	None	<u>Basin Water Chemistry Control</u> program (limits on fissile material - alpha emitters), <u>Deionizer Resin Replacement/Regeneration Program</u> , <u>Radiological Protection Program</u>	BEU 2	Radiation Monitoring	Shielding capability of the decontamination cell and RBOF structure and basin water, rack configuration, RBOF's ventilation and exhaust system	Radiological Protection Program	Evacuation and shutdown of RBOF area operations and possible surrounding area shutdown	Moderate consequence. ≥ 25 rem to the onsite worker.	13
405	E-6	Failure between interfaces	No additional scenarios identified than previously analyzed for other Event Category No. 6	----	----	----	----	----	----	----	----	N/A
406	E-7	Release of radioactive waste to soil during transfer to General Purpose (GP) Evaporator or waste tank farm.	Break in transfer line (Known leak in shroud)	None	Preventive maintenance	A	Monitors, verbal, non receipt or partial receipt at GP or tank farm	<u>Transfer Line Configuration</u>	<u>Waste Management Program</u> (waste acceptance criteria set by receiving facility)	Shutdown of affected activities, property damage,	Negligible consequence. < 0.5 rem or only a minor injury to the onsite worker.	10

Table 3.3-15 Hazard Evaluation - Waste Cell, Decontamination Cell, and RRF Cell (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
407	E-7	Transfer error to GP Evaporator at 211-H resulting in a release of radioactive waste to environment.	High-level waste sent to GP Evaporator instead of waste farm, worker error, inadequate procedure	Labeled valves and controls	NOPs	A I	Monitors, samples, non receipt or partial receipt at GP Evaporator	None	Waste Management Program (waste acceptance criteria set by receiving facility)	Shutdown of affected activities, possible contamination at Outside Facilities	Negligible consequence, < 0.5 rem or only a minor injury to the onsite worker.	10	
408	E-8	Maintenance activities	No additional scenarios identified	----	----	----	----	---	----	----	----	N/A	
409	E-9	Extrinsic Hazards	No additional scenarios identified	----	----	----	----	----	----	----	----	N/A	
410	E-10	Natural Phenomena	No additional scenarios identified	----	----	----	----	----	----	----	----	N/A	

Table 3.3-16 Hazard Evaluation - RBOF Control Room, RC&HP Office, Locker Room, Lunch Room, Office Expansion, Electrical Room, Roof Area and Basement

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
450	E-1	Ignition of combustibles in office area fire	Improper disposal of smoking materials, cooking fire, electrical short, unknown ignition source, worker error, improper maintenance operation	None	<u>Fire Protection Program</u> (Restriction of ignition sources and combustible materials), <u>8Q Safety Program</u>	A 1	Smell and Visual (Fire detectors are not installed)	Facility sections separated by partitions and shield walls, fire extinguishers, emergency exits	<u>Fire Protection Program, Event Response Procedures</u>	Shutdown of affected area operations, water damage to other equipment in the area, property damage	Low consequence. Standard Industrial Hazard.	7	
451	E-1	Explosion or decompression of gas cylinder located between lunch room and HP office	Unknown high heat source, holding chain around valve causing valve damage	<u>Cylinder design and safety valves on compressed gas cylinders</u>	<u>Fire Protection Program</u>	A 3	Audible, smell, visual	Facility sections separated by partitions and shield walls, fire extinguishers, emergency exits	<u>Fire Protection Program, Event Response Procedures</u>	Shutdown of affected area operations, water damage to other equipment in the area, property damage	Low consequence. Standard Industrial Hazard.	7	
452	E-2	No additional scenarios identified than previously analyzed for other Event Category No. 2.	----	----	----	----	----	----	----	----	----	N/A	
453	E-3	No additional scenarios identified than previously analyzed for Event Category No. 3.	----	----	----	----	----	----	----	----	----	N/A	

Table 3.3-16 Hazard Evaluation - RBOF Control Room, RC&HP Office, Locker Room, Lunch Room, Office Expansion, Electrical Room, Roof Area and Basement (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Postulated event occurrence			Risk Rank
				Design	Administrative			Design	Administrative	Consequence level	
454	E-4	No additional scenarios identified than previously analyzed for Event Category No. 4.	----	----	----	----	----	----	----	----	N/A
455	E-5	Criticality	None identified.	----	----	----	----	----	----	----	N/A
456	E-6	No additional scenarios identified than previously analyzed for other Event Category No. 6.	----	----	----	----	----	----	----	----	N/A
457	E-7	No additional scenarios identified than previously analyzed for other Event Category No. 7.	----	----	----	----	----	----	----	----	N/A
458	E-8	No additional scenarios identified than previously analyzed for other Event Category No. 8.	----	----	----	----	----	----	----	----	N/A
459	E-9	No additional scenarios identified than previously analyzed for other Event Category No. 9.	----	----	----	----	----	----	----	----	N/A

Table 3.3-16 Hazard Evaluation - RBOF Control Room, RC&HP Office, Locker Room, Lunch Room, Office Expansion, Electrical Room, Roof Area and Basement (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
460	E-10	No additional scenarios identified than previously analyzed for other Event Category no. 10.	----	----	----	----	----	----	----	----	----	N/A	

Table 3.3-17 Hazard Evaluation - Building 244-1H

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
(2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence													
Event No.	Event Category	Postulated Event Description	Causes	Prevention features		Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative			Design	Administrative	Impact on other systems	Health and Safety		
501	E-1	Ignition of combustibles resulting in area fire. (Hazardous chemicals: oxalic acid, potassium tetraborate, and sodium dichromate create irritating and toxic fumes when heated.	Electrical short, unknown ignition source, nearby vehicle fire, lightning strike causing fire.	None	None	A	Smell and visual	Building fire suppression, fire extinguishers, emergency exits	Fire Protection Program, Event Response Procedures, Hazardous Material Inventory Program, 8Q Safety Program	Shutdown of affected area operations, water or chemical damage to other equipment in the area, property damage	Low Consequence. Standard Industrial Hazard.	7	
502	E-2	No additional scenarios identified than previously analyzed Event Category No. 2.	----	----	----	----	----	----	----	----	----	N/A	
503	E-3	No additional scenarios identified than previously analyzed Event Category No. 3.	----	----	----	----	----	----	----	----	----	N/A	
504	E-4	No additional scenarios identified than previously analyzed Event Category No. 4.	----	----	----	----	----	----	----	----	----	N/A	
505	E-5	Criticality	Not applicable.	----	----	----	----	----	----	----	----	N/A	
506	E-6	No additional scenarios identified than previously analyzed Event Category No. 6.	----	----	----	----	----	----	----	----	----	N/A	

Table 3.3-17 Hazard Evaluation - Building 244-1H (Continued)

Notes: (1) Refer to last page for identification of Event Categories and Frequency Source.
 (2) Differences in facility construction and layout are noted, as appropriate.

Postulated event occurrence														
Event No.	Event Category	Postulated Event Description	Causes	Prevention features			Freq. Level & Source	Method of detection	Mitigation features			Consequence level		Risk Rank
				Design	Administrative				Design	Administrative		Impact on other systems	Health and Safety	
507	E-7	No additional scenarios identified than previously analyzed Event Category No. 7.	----	----	----	----	----	----	----	----	----	----	----	N/A
508	E-8	No additional scenarios identified than previously analyzed Event Category No. 8.	----	----	----	----	----	----	----	----	----	----	----	N/A
509	E-9	No additional scenarios identified than previously analyzed Event Category No. 9.	----	----	----	----	----	----	----	----	----	----	----	N/A
510	E-10	No additional scenarios identified than previously analyzed Event Category no. 10.	----	----	----	----	----	----	----	----	----	----	----	N/A

Event Categories		Code	Frequency Source
• E-1 - Fire/Explosion	• E-6 - Interfaces	1	SEPR Data Bank (200-Area Fault Tree Data Bank) Review
• E-2 - Loss of Containment	• E-7 - Waste Disposal	2	RBOF Systems Analysis and/or Safety Analysis
• E-3 - Loss of Confinement	• E-8 - Maintenance	3	Rare Event Theory method of frequency distribution
• E-4 - Injury to Workers	• E-9 - Extrinsic Hazards	4	UCRL-15910/ DOE-STD-1020-94, Change 1, 1996
• E-5 - Nuclear Criticality	• E-10 - Natural Phenomena	5	Fire Hazards Analysis, F-FHA-H-00014, Rev. 0

Underlined preventive and/or mitigative features credited in determining frequency and/or consequence.

Table 3.3-18 Summary of Process-Induced Criticality Facility Worker Doses

Dose Component			
Direct Dose	Water level	Dose (rem)	
	Minimum water level	12.0	
	Credible water level	0.6	
Inhalation & Immersion Dose	Criticality scenario	Dose Limit (rem)	Stay Time for Worker to Receive Dose Limit
	Instantaneous release	25.0	3.9 minutes
		100.0	15.8 minutes
	Long, slow release, undetected by NIMs with basin area ventilation:		
		Bounding (worst) case	~15 minutes
		25.0 100.0	~40 minutes
	Best-estimate case	25.0	2.5 hours
		100.0	>8 hours
	Long, slow release, undetected by NIMs with full facility ventilation:		
		Bounding (worst) case	15 minutes
		25.0 100.0	40 minutes
	Best-estimate case	25.0	2.5 hours
		100.0	>8 hours
	Long, slow release, undetected without ventilation		
		Bounding (worst) case	13.9 minutes
		25.0 100.0	27.8 minutes
	Best-estimate case	25.0	1 hour
		100.0	2 hours

Table 3.3-19 Summary of Chemical Release Air Concentrations

Chemical	Analyzed Inventory (lb. of solution)	Offsite Concentration ^a	Onsite Concentration (100 m) ^b	ERPG-3 Limit	ERPG-2 Limit
50% Nitric Acid	11,972	0.40 ppm	1.72 ppm	30 ppm	15 ppm
50% Sodium Hydroxide	13,837	^c	6.0E-03 mg/m ³	100 mg/m ³	40 mg/m ³
30% Sodium Nitrite	32,230	^c	6.0E-03 mg/m ³	500 mg/m ³	50 mg/m ³

a Concentrations documented in S-CLC-H-00304 (Ref. 186)

b Concentrations documented in S-EHA-H-00003 (Ref. 103)

c Values for offsite air concentrations were not calculated if the onsite values were less than ERPG-2 limits

Table 3.4-1 Criticality Frequency and Fission Yield Estimates for Metal/Oxide Systems

Criticality Class	Process Error Induced (per year)	High Winds Induced (per year)	Seismic Event Induced (per year)
Frequency (year ⁻¹) Estimated from Fault Tree Analysis; (Ref. Section)	6.64E-04 (3.4.2.3)	4.01E-05 (3.4.2.4)	2.55E-08 (3.4.2.5)
Conditional Probability for Event Frequency of 10 ⁻⁶ y ⁻¹ Fission yield	1.51E-03 1E+18; B Data Set; Exponential Fit	2.49E-02 7E+18; C Data Set; Log Normal Fit	N/A (Beyond Design Basis)
Fission Yield Levels Taken from the Distributions	Exceedance Probability, for Yield Level in First Column.	Exceedance Probability, for Yield Level in First Column.	Exceedance Probability, for Yield Level in First Column.
1.0E+17	0.53	0.54	0.54
5.0E+17	0.05	0.25	0.25
1.0E+18	0.0016	0.15	0.15
5.0E+18	~1.0E-14	0.035	0.035
1.0E+19	<< 1.0E-14	0.016	0.016
7.0E+19	-	0.0010	0.0010
1.0E+20	-	0.00052	0.00052
Fission Yield Levels Taken from the Distributions	Exceedance Frequency, for Yield Level in First Column. ^a	Exceedance Frequency, for Yield Level in First Column. ^a	Exceedance Frequency, for Yield Level in First Column. ^a
1.0E+17	3.7E-04	2.1E-05	1.4E-08
5.0E+17	3.5E-05	9.8E-06	6.4E-09
1.0E+18	1.1E-06	5.9E-06	3.8E-09
5.0E+18	<1E-17	1.4E-06	8.9E-10
1.0E+19	-	6.3E-07	4.1E-10
7.0E+19	-	3.9E-08	2.6E-11
1.0E+20	-	2.0E-08	1.3E-11
Evaluation Basis Criticality Fission Yield Based on 95th Exceedance Probability	5.0E+17	5.0E+18	5.0E+18
Dataset Basis	B	C	C

^a Sequence Frequency * Conditional Probability of Exceedance for Specific Yield Level

Table 3.4-2 Overall Release Fractions From Spent Fuel (Ref. 75)

Group Number	Group Name	Representative Element	Overall Release Fraction
1	Noble Gases	Xe	5.0E-01
2	Alkali Metals	Cs	1.0E-04
3	Alkali Earths	Ba	1.5E-05
4	Halogens	I	5.0E-02
5	Chalogens	Te	7.0E-02
6	Platinoids	Ru	2.0E-03
7	Transition Metals	Mo	1.5E-05
8	Tetravalent	Ce	2.0E-07
9	Trivalent	La	3.0E-07
10	Main Group I	Cd	2.0E-06
11	Main Group II	Sn	2.0E-06
12	Boron	B	3.0E-07

Table 3.4-3 MAR and Release Fractions for a Process-Induced Criticality Event (Ref. 94, 134)

Isotope Type	Material At Risk (MAR) 1.0E+20 Fissions (Ci)	Material At Risk (MAR) 5.0E+17 Fission (Ci)	Airborne Release Fraction (ARF)	Leak Path Factor (LPF)
Halogens	9.25E+05	4.63E+03	5E-02	1E+00
Noble Gases	1.51E+06	7.55E+03	5E-01	1E+00
Particulates				
Group 2	6.01E+03	3.01E+01	2E-01	5E-04
Group 3	5.07E+03	2.54E+01	3E-02	5E-04
Group 4	0.0E+00	0.0E+00	3E-02	5E-04
Group 5	1.23E+03	6.15E+00	7E-02	5E-04
Group 6	5.23E-01	2.62E-03	2E-03	5E-04
Group 7	1.99E+03	9.95E+00	3E-02	5E-04
Group 8	1.93E+01	9.65E-02	4E-04	5E-04
Group 9	2.19E+01	1.10E-01	5E-04	5E-04
Group 10	6.69E+00	3.35E-02	4E-03	5E-04
Group 11	2.02E-02	1.01E-04	4E-03	5E-04
Others	2.50E-02	1.25E-04	1E+00	5E-04

Table 3.4-4 Fuel Melting Contribution to the Source Term Factors for the Process-Induced Criticality

Isotope	RFA (Ci)	Number of RFAs	MAR (Ci)	DR x ARF	LPF	Source Term (Ci)
H-3	5.16E+01	3.71E+00	1.91E+02	5.00E-01	1.00E+00	9.57E+01
Kr-85	1.05E+03	3.71E+00	3.90E+03	5.00E-01	1.00E+00	1.95E+03
Sr-89	4.92E+01	3.71E+00	1.83E+02	3.00E-02	5.00E-04	2.74E-03
Sr-90	8.08E+03	3.71E+00	3.00E+04	3.00E-02	5.00E-04	4.50E-01
Y-90	8.08E+03	3.71E+00	3.00E+04	6.00E-04	5.00E-04	8.99E-03
Y-91	2.13E+02	3.71E+00	7.90E+02	6.00E-04	5.00E-04	2.37E-04
Zr-95	4.54E+02	3.71E+00	1.68E+03	4.00E-04	5.00E-04	3.37E-04
Nb-95	1.01E+03	3.71E+00	3.75E+03	3.00E-02	5.00E-04	5.62E-02
Nb-95M	3.37E+00	3.71E+00	1.25E+01	3.00E-02	5.00E-04	1.88E-04
Tc-99	1.03E+00	3.71E+00	3.82E+00	3.00E-02	5.00E-04	5.73E-05
Ru-103	2.17E+00	3.71E+00	8.05E+00	2.00E-03	1.00E+00	1.61E-02
Ru-106	2.11E+04	3.71E+00	7.83E+04	2.00E-03	1.00E+00	1.57E+02
Rh-103m	1.96E+00	3.71E+00	7.27E+00	2.00E-03	1.00E+00	1.45E-02
Rh-106	2.11E+04	3.71E+00	7.83E+04	2.00E-03	1.00E+00	1.57E+02
Ag-110	2.32E+00	3.71E+00	8.61E+00	4.00E-03	5.00E-04	1.72E-05
Ag-110M	1.74E+02	3.71E+00	6.46E+02	4.00E-03	5.00E-04	1.29E-03
Cd-113M	6.95E+00	3.71E+00	2.58E+01	4.00E-03	5.00E-04	5.16E-05
Sn-119M	3.93E+00	3.71E+00	1.46E+01	4.00E-03	5.00E-04	2.92E-05
Sn-123	1.45E+01	3.71E+00	5.38E+01	4.00E-03	5.00E-04	1.08E-04
Sb-125	8.70E+02	3.71E+00	3.23E+03	4.00E-03	5.00E-04	6.46E-03
Te-125M	2.12E+02	3.71E+00	7.87E+02	7.00E-02	1.00E+00	5.51E+01
Te-127	3.47E+01	3.71E+00	1.29E+02	7.00E-02	1.00E+00	9.01E+00
Te-127M	3.54E+01	3.71E+00	1.31E+02	7.00E-02	1.00E+00	9.19E+00
Te-129	1.20E-03	3.71E+00	4.45E-03	7.00E-02	1.00E+00	3.12E-04
Te-129M	1.85E-03	3.71E+00	6.86E-03	7.00E-02	1.00E+00	4.80E-04
Cs-134	1.03E+04	3.71E+00	3.82E+04	7.00E-02	1.00E+00	3.82E+00
Cs-137	9.28E+03	3.71E+00	3.44E+04	7.00E-02	1.00E+00	3.44E+00
Ba-137M	8.78E+03	3.71E+00	3.26E+04	7.00E-02	1.00E+00	4.89E-01
Ce-141	6.46E-02	3.71E+00	2.40E-01	4.00E-04	5.00E-04	4.79E-08
Ce-144	4.78E+04	3.71E+00	1.77E+05	4.00E-04	5.00E-04	3.55E-02
Pr-144	4.78E+04	3.71E+00	1.77E+05	6.00E-04	5.00E-04	5.32E-02
Pr-144M	5.74E+02	3.71E+00	2.13E+03	6.00E-04	5.00E-04	6.39E-04

Table 3.4-4 Fuel Melting Contribution to the Source Term Factors for the Process-Induced Criticality (continued)

Isotope	RFA (Ci)	Number of RFAs	MAR (Ci)	DR x ARF	LPF	Source Term (Ci)
Pm-147	1.88E+04	3.71E+00	6.97E+04	6.00E-04	5.00E-04	2.09E-02
Pm-148M	8.93E-03	3.71E+00	3.31E-02	6.00E-04	5.00E-04	9.94E-09
Sm-151	6.94E+01	3.71E+00	2.57E+02	6.00E-04	5.00E-04	7.72E-05
Eu-154	7.27E+02	3.71E+00	2.70E+03	6.00E-04	5.00E-04	8.09E-04
Eu-155	3.81E+02	3.71E+00	1.41E+03	6.00E-04	5.00E-04	4.24E-04
Tl-208	8.46E+00	3.71E+00	3.14E+01	4.00E-03	5.00E-04	6.28E-05
Pb-209	8.74E-03	3.71E+00	3.24E-02	4.00E-03	5.00E-04	6.49E-08
Pb-211	1.66E-02	3.71E+00	6.16E-02	4.00E-03	5.00E-04	1.23E-07
Pb-212	2.36E+01	3.71E+00	8.76E+01	4.00E-03	5.00E-04	1.75E-04
Bi-211	1.66E-02	3.71E+00	6.16E-02	4.00E-03	5.00E-04	1.23E-07
Bi-212	2.36E+01	3.71E+00	8.76E+01	4.00E-03	5.00E-04	1.75E-04
Bi-213	8.74E-03	3.71E+00	3.24E-02	4.00E-03	5.00E-04	6.49E-08
Po-212	1.51E+01	3.71E+00	5.60E+01	7.00E-02	1.00E+00	3.92E+00
Po-213	8.55E-03	3.71E+00	3.17E-02	7.00E-02	1.00E+00	2.22E-03
Po-215	1.66E-02	3.71E+00	6.16E-02	7.00E-02	1.00E+00	4.31E-03
Po-216	2.36E+01	3.71E+00	8.76E+01	7.00E-02	1.00E+00	6.13E+00
At-217	8.74E-03	3.71E+00	3.24E-02	5.00E-02	1.00E+00	1.62E-03
Rn-219	1.66E-02	3.71E+00	6.16E-02	5.00E-01	1.00E+00	3.08E-02
Rn-220	2.36E+01	3.71E+00	8.76E+01	5.00E-01	1.00E+00	4.38E+01
Fr-221	8.74E-03	3.71E+00	3.24E-02	2.00E-01	5.00E-04	3.24E-06
Ra-223	1.66E-02	3.71E+00	6.16E-02	3.00E-02	5.00E-04	9.24E-07
Ra-224	2.36E+01	3.71E+00	8.76E+01	3.00E-02	5.00E-04	1.31E-03
Ra-225	8.74E-03	3.71E+00	3.24E-02	3.00E-02	5.00E-04	4.86E-07
Ac-225	8.74E-03	3.71E+00	3.24E-02	6.00E-04	5.00E-04	9.73E-09
Ac-227	1.71E-02	3.71E+00	6.34E-02	6.00E-04	5.00E-04	1.90E-08
Th-227	1.64E-02	3.71E+00	6.08E-02	4.00E-04	5.00E-04	1.22E-08
Th-228	2.35E+01	3.71E+00	8.72E+01	4.00E-04	5.00E-04	1.74E-05
Th-229	8.74E-03	3.71E+00	3.24E-02	4.00E-04	5.00E-04	6.49E-09
Th-231	1.14E-02	3.71E+00	4.23E-02	4.00E-04	5.00E-04	8.46E-09
Th-232	1.72E-02	3.71E+00	6.38E-02	4.00E-04	5.00E-04	1.28E-08
Th-234	2.16E-04	3.71E+00	8.01E-04	4.00E-04	5.00E-04	1.60E-10
Pa-231	2.28E-01	3.71E+00	8.46E-01	4.00E-04	5.00E-04	1.69E-07

Table 3.4-4 Fuel Melting Contribution to the Source Term Factors for the Process-Induced Criticality (continued)

Isotope	RFA (Ci)	Number of RFAs	MAR (Ci)	DR x ARF	LPF	Source Term (Ci)
Pa-233	8.59E-02	3.71E+00	3.19E-01	4.00E-04	5.00E-04	6.37E-08
Pa-234M	2.16E-04	3.71E+00	8.01E-04	4.00E-04	5.00E-04	1.60E-10
U-232	4.09E+01	3.71E+00	1.52E+02	4.00E-04	5.00E-04	3.03E-05
U-233	3.93E+01	3.71E+00	1.46E+02	4.00E-04	5.00E-04	2.92E-05
U-234	1.92E+00	3.71E+00	7.12E+00	4.00E-04	5.00E-04	1.42E-06
U-235	1.14E-02	3.71E+00	4.23E-02	4.00E-04	5.00E-04	8.46E-09
U-236	3.29E-02	3.71E+00	1.22E-01	4.00E-04	5.00E-04	2.44E-08
U-237	2.59E-01	3.71E+00	9.61E-01	4.00E-04	5.00E-04	1.92E-07
U-238	8.42E-02	3.71E+00	3.12E-01	4.00E-04	5.00E-04	6.25E-08
Np-237	8.81E-03	3.71E+00	3.27E-02	4.00E-04	5.00E-04	6.54E-09
Np-239	9.62E+00	3.71E+00	3.57E+01	4.00E-04	5.00E-04	7.14E-06
Pu-236	1.12E+02	3.71E+00	4.16E+02	4.00E-04	5.00E-04	8.31E-05
Pu-238	5.19E+01	3.71E+00	1.93E+02	4.00E-04	5.00E-04	3.85E-05
Pu-239	5.80E+01	3.71E+00	2.15E+02	4.00E-04	5.00E-04	4.30E-05
Pu-240	9.78E+03	3.71E+00	3.63E+04	4.00E-04	5.00E-04	7.26E-03
Pu-241	1.06E+04	3.71E+00	3.93E+04	4.00E-04	5.00E-04	7.87E-03
Am-241	5.17E+01	3.71E+00	1.92E+02	6.00E-04	5.00E-04	5.75E-05
Am-242m	3.41E-01	3.71E+00	1.27E+00	6.00E-04	5.00E-04	3.80E-07
Am-242	3.40E-01	3.71E+00	1.26E+00	6.00E-04	5.00E-04	3.78E-07
Am-243	9.62E+00	3.71E+00	3.57E+01	6.00E-04	5.00E-04	1.07E-05
Cm-242	4.90E+02	3.71E+00	1.82E+03	6.00E-04	5.00E-04	5.45E-04
Cm-243	4.90E+00	3.71E+00	1.82E+01	6.00E-04	5.00E-04	5.45E-06
Cm-244	2.75E+03	3.71E+00	1.02E+04	6.00E-04	5.00E-04	3.06E-03
Cm-246	2.15E-01	3.71E+00	7.98E-01	6.00E-04	5.00E-04	2.39E-07
	2.31E+05					2.50E+03

Table 3.4-5 Fission Products after 10^{20} Fissions

Isotope	Time Following Criticality Termination		
	60.0S	600.0S	3600.0S
ZN73	1.96E+02	2.37E-05	0.00E+00
ZN74	3.35E+02	6.51E+00	2.03E-09
GA74	3.75E+01	5.35E+01	7.63E-01
GA75	5.69E+02	2.16E+01	2.59E-07
GE75	4.37E+00	1.62E+01	1.10E+01
GA76	1.96E+03	1.97E-03	0.00E+00
GA77	1.32E+03	4.12E-10	0.00E+00
GE77	2.45E+00	3.66E+00	3.47E+00
GE77M	3.89E+03	4.29E+00	1.00E-16
AS77	8.74E-01	2.18E+00	2.20E+00
GE78	1.66E+02	1.55E+02	1.04E+02
AS78	2.21E+00	1.27E+01	4.87E+01
GE79	1.12E+04	1.85E+00	1.85E-21
AS79	1.49E+03	1.23E+03	2.61E+01
SE79M	1.55E+02	1.33E+03	4.62E+01
GE80	1.91E+04	3.21E-03	0.00E+00
AS80	3.48E+04	1.03E-02	0.00E+00
GE81	5.20E+03	4.18E-13	0.00E+00
AS81	4.83E+04	4.21E-01	0.00E+00
SE81	2.57E+03	2.90E+03	4.57E+02
SE81M	2.42E+01	2.17E+01	1.18E+01
AS82	3.34E+04	6.07E-04	0.00E+00
AS82M	4.77E+03	1.49E-09	0.00E+00
AS83	2.89E+04	2.63E-08	0.00E+00
SE83	3.14E+03	2.46E+03	5.27E+02
SE83M	6.00E+04	3.07E+02	3.85E-11
BR83	3.00E+02	9.11E+02	9.79E+02
KR83M	9.01E-01	4.38E+01	2.99E+02
AS84	1.06E+03	0.00E+00	0.00E+00
SE84	8.32E+04	1.26E+04	3.45E-01
BR84	2.15E+03	8.23E+03	3.26E+03
BR84M	8.21E+02	2.90E+02	9.00E-01
SE85	1.15E+05	7.81E+00	0.00E+00
SE85M	5.40E+04	1.50E-04	0.00E+00
BR85	9.72E+04	1.56E+04	8.77E-02
KR85M	1.88E+02	1.47E+03	1.44E+03
SE86	1.27E+05	2.05E-05	0.00E+00
BR86	1.78E+05	2.28E+02	8.67E-15
BR86M	8.75E+04	1.41E-05	0.00E+00
SE87	1.94E+03	0.00E+00	0.00E+00
BR87	3.71E+05	4.53E+02	2.96E-14
KR87	6.56E+03	1.02E+04	6.48E+03
BR88	2.24E+05	2.39E-05	0.00E+00
KR88	6.59E+03	6.68E+03	5.45E+03

Isotope	Time Following Criticality Termination		
	60.0S	600.0S	3600.0S
RB89	2.20E+04	6.47E+04	7.98E+03
SR89	1.50E-01	4.84E+00	1.95E+01
KR90	7.66E+05	7.15E+00	0.00E+00
RB90	3.65E+05	4.82E+04	5.81E-01
RB90M	5.35E+04	1.57E+04	4.95E+00
KR91	5.33E+04	1.10E-14	0.00E+00
RB91	8.96E+05	1.46E+03	4.42E-13
SR91	1.68E+03	3.19E+03	3.00E+03
Y91M	9.82E+00	2.13E+02	1.00E+03
RB92	2.07E+03	0.00E+00	0.00E+00
SR92	1.16E+04	1.11E+04	9.00E+03
Y92	6.95E+01	3.96E+02	1.84E+03
RB93	8.41E+03	0.00E+00	0.00E+00
SR93	2.49E+05	1.08E+05	1.07E+03
Y93	3.32E+02	2.06E+03	3.22E+03
SR94	8.69E+05	6.15E+03	6.97E-09
Y94	4.57E+04	7.68E+04	1.26E+04
SR95	8.20E+05	4.59E-01	0.00E+00
Y95	1.55E+05	1.05E+05	3.87E+03
ZR95	9.25E-01	1.06E+01	2.22E+01
SR96	5.78E+02	0.00E+00	0.00E+00
Y96	6.31E+05	4.19E+04	1.20E-02
ZR97	1.87E+03	1.86E+03	1.80E+03
NB97	3.15E+02	4.82E+02	9.91E+02
NB97M	4.80E+03	1.77E+03	1.70E+03
ZR98	9.74E+05	5.55E+00	0.00E+00
NB98	1.07E+06	6.11E+00	0.00E+00
NB98M	1.55E+02	1.37E+02	7.00E+01
NB99	5.08E+05	2.18E-06	0.00E+00
NB99M	1.22E+04	1.11E+03	1.81E-03
MO99	5.09E+02	5.46E+02	5.42E+02
TC99M	5.98E-01	8.74E+00	5.15E+01
ZR100	4.30E+04	0.00E+00	0.00E+00
NB100	3.25E+04	0.00E+00	0.00E+00
NB100M	3.26E+04	0.00E+00	0.00E+00
NB101	6.32E+04	0.00E+00	0.00E+00
MO101	1.73E+05	1.13E+05	1.06E+04
TC101	6.95E+03	5.39E+04	2.96E+04
ZR102	2.99E+05	6.24E-01	0.00E+00
NB102	3.34E+05	6.97E-01	0.00E+00
MO102	1.13E+05	7.28E+04	3.21E+03
TC102	1.13E+05	7.34E+04	3.23E+03
TC102M	9.88E+01	2.35E+01	8.16E-03
NB103	2.04E+05	8.61E-06	0.00E+00

Table 3.4-5 Fission Products after 10^{20} Fissions (continued)

Isotope	Time Following Criticality Termination		
	60.0S	600.0S	3600.0S
RB88	7.41E+02	2.53E+03	5.39E+03
BR89	6.87E+02	0.00E+00	0.00E+00
KR89	3.63E+05	5.07E+04	9.06E-01
RH103M	9.01E-03	1.67E+00	8.99E+00
MO104	2.88E+05	5.83E+03	2.28E-06
TC104	1.48E+04	2.96E+04	4.49E+03
MO105	1.84E+05	1.80E+02	3.39E-15
TC105	2.47E+04	2.20E+04	2.89E+02
RU105	3.98E+01	8.08E+02	1.30E+03
RH105	5.21E+00	6.55E+00	2.65E+01
RH105M	3.17E+00	2.26E+02	3.64E+02
MO106	1.12E+04	9.73E-15	0.00E+00
TC106	1.27E+05	5.28E+00	0.00E+00
MO107	1.28E+03	0.00E+00	0.00E+00
TC107	7.38E+04	1.84E-01	0.00E+00
RU107	1.98E+04	6.66E+03	1.74E+00
RH107	3.62E+02	3.97E+03	1.13E+03
TC108	3.07E+02	0.00E+00	0.00E+00
RU108	1.45E+04	3.63E+03	1.64E+00
RH108	1.39E+04	3.87E+03	1.75E+00
TC109	1.58E+04	1.03E+01	2.01E-17
RU109	2.55E+04	3.21E+01	6.41E-17
RH109	7.67E+03	1.36E+03	1.44E-07
RH109M	8.48E+03	1.03E+02	1.00E-15
PD109	1.82E+00	1.32E+02	1.58E+02
PD109M	2.84E+02	5.16E+03	3.40E+00
AG109M	5.07E-01	1.32E+02	1.58E+02
RU110	9.82E+03	6.80E-07	0.00E+00
RH110	2.70E+04	9.70E-02	0.00E+00
RU111	6.10E+03	1.75E-07	0.00E+00
RH111	1.57E+04	4.63E+01	2.14E-13
PD111	4.63E+02	1.01E+03	2.10E+02
AG111	2.49E-03	5.03E-01	2.23E+00
AG111M	1.13E+02	1.04E+03	2.19E+02
PD112	1.99E+01	1.98E+01	1.92E+01
AG112	3.87E+00	4.39E+00	6.93E+00
PD113	1.04E+04	1.63E+02	1.51E-08
AG113	2.56E+01	6.80E+01	6.16E+01
AG113M	6.06E+02	4.97E+01	5.64E-09
PD114	7.40E+03	5.50E+02	2.94E-04
AG114	7.64E+03	5.68E+02	3.04E-04
PD115	1.25E+04	6.59E-01	0.00E+00
AG115	5.24E+02	6.02E+02	1.06E+02
AG115M	4.44E+03	3.22E-01	0.00E+00

Isotope	Time Following Criticality Termination		
	60.0S	600.0S	3600.0S
MO103	5.99E+05	1.31E+03	1.17E-12
TC103	3.55E+05	8.07E+03	6.99E-12
RU103	2.37E+00	1.94E+01	1.95E+01
CD117	6.29E+01	9.47E+01	7.60E+01
CD117M	3.28E+01	3.85E+01	3.25E+01
IN117	3.19E-01	4.10E+00	2.08E+01
IN117M	3.55E-01	5.57E+00	2.84E+01
CD118	4.72E+02	4.17E+02	2.09E+02
IN118	4.72E+02	4.17E+02	2.10E+02
AG119	1.97E+02	0.00E+00	0.00E+00
CD119	1.24E+03	6.38E+02	1.60E+01
CD119M	3.21E+03	4.57E+02	9.04E-03
IN119	3.93E+02	4.93E+02	9.13E+00
IN119M	1.04E+02	5.30E+02	1.69E+02
CD120	1.26E+04	7.94E+00	1.33E-17
IN120	5.64E+03	2.18E+01	5.26E-17
IN120M	6.70E+03	4.23E+00	7.05E-18
CD121	4.39E+03	8.76E-10	0.00E+00
IN121	1.49E+04	2.80E-02	0.00E+00
IN121M	1.28E+03	2.01E+02	5.52E-03
SN121	8.27E+00	1.53E+01	1.54E+01
CD122	1.19E+02	0.00E+00	0.00E+00
IN122	4.45E+03	2.55E-13	0.00E+00
CD123	9.17E+02	4.17E-17	0.00E+00
IN123	2.16E+03	1.11E-16	0.00E+00
IN123M	5.00E+03	2.07E+00	3.17E-19
SN123M	4.93E+02	4.72E+02	1.99E+02
CD124	4.39E+03	1.50E-06	0.00E+00
IN124	5.40E+03	1.84E-06	0.00E+00
IN125M	2.02E+03	5.73E-11	0.00E+00
SN125M	2.77E+03	1.46E+03	3.83E+01
SB126M	3.77E+01	2.72E+01	4.38E+00
SN127	5.68E+02	5.41E+02	4.11E+02
SN127M	7.52E+03	1.66E+03	3.79E-01
SB127	2.81E+00	7.79E+00	1.19E+01
SN128	3.35E+03	3.01E+03	1.67E+03
SB128	2.88E+01	2.84E+01	2.67E+01
SB128M	1.60E+03	2.30E+03	1.98E+03
SN129	1.33E+04	5.81E+03	5.72E+01
SN129M	3.55E+04	2.92E+03	2.79E-03
SB129	5.31E+02	1.04E+03	1.09E+03
TE129	2.12E+02	2.58E+02	5.41E+02
SN130	9.14E+04	1.71E+04	1.54E+00
SB130	2.33E+03	2.00E+03	8.40E+02

Table 3.4-5 Fission products after 10^{20} Fissions (continued)

Isotope	Time Following Criticality Termination		
	60.0S	600.0S	3600.0S
CD115	7.39E-01	2.85E+00	5.83E+00
PD116	4.63E+03	1.13E-08	0.00E+00
AG116	3.49E+03	3.61E+02	8.75E-04
AG116M	5.92E+03	2.20E-08	0.00E+00
AG117	5.69E+03	3.42E+01	1.57E-11
AG117M	3.53E+02	0.00E+00	0.00E+00
SN132	1.12E+05	9.63E+00	0.00E+00
SB132	1.49E+05	1.98E+04	8.34E-02
SB132M	7.74E+04	1.75E+04	4.57E+00
TE132	1.43E+02	2.94E+02	3.19E+02
I132	1.24E+02	1.30E+02	1.72E+02
SB133	2.27E+05	1.69E+04	9.03E-03
TE133	5.93E+04	6.47E+04	5.14E+03
TE133M	1.25E+04	1.13E+04	6.06E+03
I133	1.19E+02	5.18E+02	1.36E+03
I133M	2.96E+03	2.57E-15	0.00E+00
XE133	1.36E-01	3.92E-01	5.20E+00
SB134	1.80E+04	3.00E-11	0.00E+00
SB134M	1.58E+04	1.01E-11	0.00E+00
TE134	4.82E+04	4.17E+04	1.82E+04
I134	5.23E+03	1.20E+04	1.96E+04
I134M	4.47E+04	8.28E+03	7.08E-01
TE135	4.05E+05	1.38E-03	0.00E+00
I135	4.56E+03	4.82E+03	4.41E+03
XE135	9.24E+01	1.84E+02	5.22E+02
XE135M	4.77E+03	3.42E+03	9.82E+02
CS135M	5.97E+00	5.31E+00	2.76E+00
TE136	2.20E+05	3.99E-03	0.00E+00
I136	4.75E+05	6.04E+03	7.96E-08
I136M	2.94E+05	8.60E+01	2.01E-18
I137	4.81E+05	1.19E-01	0.00E+00
XE137	3.87E+05	8.67E+04	1.02E+01
I138	8.07E+03	0.00E+00	0.00E+00
XE138	1.33E+05	8.54E+04	7.40E+03
CS138	5.78E+03	2.46E+04	2.55E+04
CS138M	2.07E+04	2.41E+03	1.56E-02
XE139	8.06E+05	6.18E+01	0.00E+00
CS139	1.52E+05	1.10E+05	2.75E+03
BA139	1.21E+03	1.24E+04	1.71E+04
XE140	2.03E+05	2.26E-07	0.00E+00
CS140	9.52E+05	2.85E+03	1.99E-11
BA140	4.84E+01	1.06E+02	1.06E+02

Isotope	Time Following Criticality Termination		
	60.0S	600.0S	3600.0S
SB130M	2.46E+04	3.35E+04	2.35E+02
SN131	1.97E+05	5.18E+02	2.40E-12
SB131	3.03E+04	3.02E+04	6.71E+03
TE131	3.02E+03	9.17E+03	1.15E+04
TE131M	3.67E+01	4.44E+01	6.38E+01
I131	3.79E-01	3.78E+00	4.24E+01
LA144	1.11E+06	1.01E+02	0.00E+00
CE144	2.31E+00	4.19E+00	4.19E+00
PR144	3.09E+00	3.70E+00	4.17E+00
BA145	7.80E+03	0.00E+00	0.00E+00
LA145	6.59E+05	1.64E+00	0.00E+00
CE145	2.64E+05	4.89E+04	4.70E-01
PR145	3.22E+02	3.15E+03	3.23E+03
LA146	4.05E+04	1.05E-15	0.00E+00
CE146	6.78E+04	4.39E+04	3.83E+03
PR146	1.74E+03	1.38E+04	1.28E+04
LA147	3.41E+04	1.90E-12	0.00E+00
CE147	3.81E+05	1.84E+03	2.31E-10
PR147	2.60E+04	4.00E+04	2.24E+03
ND147	6.10E-01	1.82E+01	4.68E+01
CE148	2.66E+05	4.41E+01	4.39E-20
PR148	1.30E+05	1.66E+04	4.74E-03
PR149	1.13E+05	7.48E+03	2.14E-03
ND149	9.17E+02	3.10E+03	2.34E+03
PM149	1.12E-01	4.98E+00	3.50E+01
PR150	3.60E+04	2.80E-09	0.00E+00
ND151	1.08E+04	6.50E+03	3.97E+02
PM151	4.75E+00	3.56E+01	7.87E+01
CE152	1.16E+03	3.01E-09	0.00E+00
PR152	4.36E+03	7.39E-09	0.00E+00
ND152	7.58E+03	4.45E+03	2.19E+02
PM152	1.34E+03	4.61E+03	3.41E+02
PM152M	1.46E+02	6.34E+01	6.24E-01
PR153	4.83E+02	4.90E-19	0.00E+00
ND153	2.81E+04	1.10E+02	4.70E-12
PM153	5.42E+03	4.01E+03	6.58E+00
SM153	9.18E-01	1.49E+01	2.24E+01
ND154	1.11E+04	9.60E-01	0.00E+00
PM154	5.21E+03	9.54E+02	4.05E-03
PM154M	1.34E+03	4.18E+01	1.82E-07
ND155	3.80E+03	2.20E-03	0.00E+00
PM155	9.80E+03	6.83E-01	0.00E+00

Table 3.4-5 Fission Products After 10^{20} Fissions (continued)

Isotope	Time Following Criticality Termination		
	60.0S	600.0S	3600.0S
LA140	1.18E+00	1.43E+00	2.92E+00
CS141	6.63E+05	2.09E-01	0.00E+00
BA141	8.91E+04	7.43E+04	1.11E+04
LA141	2.22E+02	2.53E+03	6.63E+03
CE141	1.89E-03	1.95E-01	4.16E+00
BA142	1.56E+05	8.70E+04	3.41E+03
LA142	1.54E+03	9.10E+03	1.36E+04
BA143	3.66E+05	4.08E-07	0.00E+00
LA143	1.25E+05	8.37E+04	7.04E+03
CE143	3.46E+01	3.66E+02	8.95E+02
BA144	1.69E+05	2.82E-10	0.00E+00

Isotope	Time Following Criticality Termination		
	60.0S	600.0S	3600.0S
SM155	4.53E+02	6.10E+02	1.28E+02
ND156	8.12E+02	1.35E+00	4.91E-16
PM156	1.76E+03	1.74E+00	6.32E-16
SM156	1.18E+01	1.37E+01	1.29E+01
PM157	1.24E+03	5.04E+00	2.66E-13
SM157	4.14E+02	2.82E+02	3.72E+00
EU157	3.96E-01	3.06E+00	5.33E+00
SM158	5.61E+01	4.87E+01	2.21E+01
EU158	5.83E+00	1.17E+01	2.25E+01
EU159	2.29E+01	3.94E+01	6.37E+00
SM160	5.18E+01	1.77E+01	4.58E-02

Table 3.4-6 MAR and Release Fractions for an NPH-Induced Criticality Event (Ref. 94, 134)

Isotope Type	Material At Risk (MAR) 1E+20 Fissions (Ci)	Material At Risk (MAR) 5E+18 Fissions (Ci)	Airborne Release Fraction (ARF)	Leak Path Factor (LPF)
Halogens	9.25E+05	4.63E+04	5E-02	1E+00
Noble Gases	1.51E+06	7.55E+04	5E-01	1E+00
Particulates				
Group 2	6.01E+03	3.01E+02	2E-01	5E-04
Group 3	5.07E+03	2.54E+02	3E-02	5E-04
Group 4	0.0E+00	0.0E+00	3E-02	5E-04
Group 5	1.23E+03	6.15E+01	7E-02	5E-04
Group 6	5.23E-01	2.62E-02	2E-03	5E-04
Group 7	1.99E+03	1.00E+02	3E-02	5E-04
Group 8	1.93E+01	9.7E-01	4E-04	5E-04
Group 9	2.19E+01	1.10E+00	5E-04	5E-04
Group 10	6.69E+00	3.35E-01	4E-03	5E-04
Group 11	2.02E-02	1.01E-03	4E-03	5E-04
Others	2.50E-02	1.25E-03	1E+00	5E-04

Table 3.4-7 Fuel Melting Contribution to the Source Term for the NPH Criticality

Isotope	RFA (Ci)	No. RFAs	MAR	DR x ARF	LPF	Fuel Melt (Ci)
H-3	5.16E+01	3.71E+01	1.91E+03	5.00E-01	1.00E+00	9.57E+02
Kr-85	1.05E+03	3.71E+01	3.90E+04	5.00E-01	1.00E+00	1.95E+04
Sr-89	4.92E+01	3.71E+01	1.83E+03	3.00E-02	5.00E-04	2.74E-02
Sr-90	8.08E+03	3.71E+01	3.00E+05	3.00E-02	5.00E-04	4.50E+00
Y-90	8.08E+03	3.71E+01	3.00E+05	6.00E-04	5.00E-04	8.99E-02
Y-91	2.13E+02	3.71E+01	7.90E+03	6.00E-04	5.00E-04	2.37E-03
Zr-95	4.54E+02	3.71E+01	1.68E+04	4.00E-04	5.00E-04	3.37E-03
Nb-95	1.01E+03	3.71E+01	3.75E+04	3.00E-02	5.00E-04	5.62E-01
Nb-95M	3.37E+00	3.71E+01	1.25E+02	3.00E-02	5.00E-04	1.88E-03
Tc-99	1.03E+00	3.71E+01	3.82E+01	3.00E-02	5.00E-04	5.73E-04
Ru-103	2.17E+00	3.71E+01	8.05E+01	2.00E-03	1.00E+00	1.61E-01
Ru-106	2.11E+04	3.71E+01	7.83E+05	2.00E-03	1.00E+00	1.57E+03
Rh-103m	1.96E+00	3.71E+01	7.27E+01	2.00E-03	1.00E+00	1.45E-01
Rh-106	2.11E+04	3.71E+01	7.83E+05	2.00E-03	1.00E+00	1.57E+03
Ag-110	2.32E+00	3.71E+01	8.61E+01	4.00E-03	5.00E-04	1.72E-04
Ag-110M	1.74E+02	3.71E+01	6.46E+03	4.00E-03	5.00E-04	1.29E-02
Cd-113M	6.95E+00	3.71E+01	2.58E+02	4.00E-03	5.00E-04	5.16E-04
Sn-119M	3.93E+00	3.71E+01	1.46E+02	4.00E-03	5.00E-04	2.92E-04
Sn-123	1.45E+01	3.71E+01	5.38E+02	4.00E-03	5.00E-04	1.08E-03
Sb-125	8.70E+02	3.71E+01	3.23E+04	4.00E-03	5.00E-04	6.46E-02
Te-125M	2.12E+02	3.71E+01	7.87E+03	7.00E-02	1.00E+00	5.51E+02
Te-127	3.47E+01	3.71E+01	1.29E+03	7.00E-02	1.00E+00	9.01E+01
Te-127M	3.54E+01	3.71E+01	1.31E+03	7.00E-02	1.00E+00	9.19E+01
Te-129	1.20E-03	3.71E+01	4.45E-02	7.00E-02	1.00E+00	3.12E-03
Te-129M	1.85E-03	3.71E+01	6.86E-02	7.00E-02	1.00E+00	4.80E-03
Cs-134	1.03E+04	3.71E+01	3.82E+05	7.00E-02	1.00E+00	3.82E+01
Cs-137	9.28E+03	3.71E+01	3.44E+05	7.00E-02	1.00E+00	3.44E+01
Ba-137M	8.78E+03	3.71E+01	3.26E+05	7.00E-02	1.00E+00	4.89E+00
Ce-141	6.46E-02	3.71E+01	2.40E+00	4.00E-04	5.00E-04	4.79E-07
Ce-144	4.78E+04	3.71E+01	1.77E+06	4.00E-04	5.00E-04	3.55E-01
Pr-144	4.78E+04	3.71E+01	1.77E+06	6.00E-04	5.00E-04	5.32E-01
Pr-144M	5.74E+02	3.71E+01	2.13E+04	6.00E-04	5.00E-04	6.39E-03
Pm-147	1.88E+04	3.71E+01	6.97E+05	6.00E-04	5.00E-04	2.09E-01

Table 3.4-7 Fuel Melting Contribution to the Source Term for the NPH Criticality (continued)

Isotope	RFA (Ci)	No. RFAs	MAR	DR x ARF	LPF	Fuel Melt (Ci)
Pm-148M	8.93E-03	3.71E+01	3.31E-01	6.00E-04	5.00E-04	9.94E-08
Sm-151	6.94E+01	3.71E+01	2.57E+03	6.00E-04	5.00E-04	7.72E-04
Eu-154	7.27E+02	3.71E+01	2.70E+04	6.00E-04	5.00E-04	8.09E-03
Eu-155	3.81E+02	3.71E+01	1.41E+04	6.00E-04	5.00E-04	4.24E-03
Tl-208	8.46E+00	3.71E+01	3.14E+02	4.00E-03	5.00E-04	6.28E-04
Pb-209	8.74E-03	3.71E+01	3.24E-01	4.00E-03	5.00E-04	6.49E-07
Pb-211	1.66E-02	3.71E+01	6.16E-01	4.00E-03	5.00E-04	1.23E-06
Pb-212	2.36E+01	3.71E+01	8.76E+02	4.00E-03	5.00E-04	1.75E-03
Bi-211	1.66E-02	3.71E+01	6.16E-01	4.00E-03	5.00E-04	1.23E-06
Bi-212	2.36E+01	3.71E+01	8.76E+02	4.00E-03	5.00E-04	1.75E-03
Bi-213	8.74E-03	3.71E+01	3.24E-01	4.00E-03	5.00E-04	6.49E-07
Po-212	1.51E+01	3.71E+01	5.60E+02	7.00E-02	1.00E+00	3.92E+01
Po-213	8.55E-03	3.71E+01	3.17E-01	7.00E-02	1.00E+00	2.22E-02
Po-215	1.66E-02	3.71E+01	6.16E-01	7.00E-02	1.00E+00	4.31E-02
Po-216	2.36E+01	3.71E+01	8.76E+02	7.00E-02	1.00E+00	6.13E+01
At-217	8.74E-03	3.71E+01	3.24E-01	5.00E-02	1.00E+00	1.62E-02
Rn-219	1.66E-02	3.71E+01	6.16E-01	5.00E-01	1.00E+00	3.08E-01
Rn-220	2.36E+01	3.71E+01	8.76E+02	5.00E-01	1.00E+00	4.38E+02
Fr-221	8.74E-03	3.71E+01	3.24E-01	2.00E-01	5.00E-04	3.24E-05
Ra-223	1.66E-02	3.71E+01	6.16E-01	3.00E-02	5.00E-04	9.24E-06
Ra-224	2.36E+01	3.71E+01	8.76E+02	3.00E-02	5.00E-04	1.31E-02
Ra-225	8.74E-03	3.71E+01	3.24E-01	3.00E-02	5.00E-04	4.86E-06
Ac-225	8.74E-03	3.71E+01	3.24E-01	6.00E-04	5.00E-04	9.73E-08
Ac-227	1.71E-02	3.71E+01	6.34E-01	6.00E-04	5.00E-04	1.90E-07
Th-227	1.64E-02	3.71E+01	6.08E-01	4.00E-04	5.00E-04	1.22E-07
Th-228	2.35E+01	3.71E+01	8.72E+02	4.00E-04	5.00E-04	1.74E-04
Th-229	8.74E-03	3.71E+01	3.24E-01	4.00E-04	5.00E-04	6.49E-08
Th-231	1.14E-02	3.71E+01	4.23E-01	4.00E-04	5.00E-04	8.46E-08
Th-232	1.72E-02	3.71E+01	6.38E-01	4.00E-04	5.00E-04	1.28E-07
Th-234	2.16E-04	3.71E+01	8.01E-03	4.00E-04	5.00E-04	1.60E-09
Pa-231	2.28E-01	3.71E+01	8.46E+00	4.00E-04	5.00E-04	1.69E-06
Pa-233	8.59E-02	3.71E+01	3.19E+00	4.00E-04	5.00E-04	6.37E-07
Pa-234M	2.16E-04	3.71E+01	8.01E-03	4.00E-04	5.00E-04	1.60E-09

Table 3.4-7 Fuel Melting Contribution to the Source Term for the NPH Criticality (continued)

Isotope	RFA (Ci)	No. RFAs	MAR	DR x ARF	LPF	Fuel Melt (Ci)
U-232	4.09E+01	3.71E+01	1.52E+03	4.00E-04	5.00E-04	3.03E-04
U-233	3.93E+01	3.71E+01	1.46E+03	4.00E-04	5.00E-04	2.92E-04
U-234	1.92E+00	3.71E+01	7.12E+01	4.00E-04	5.00E-04	1.42E-05
U-235	1.14E-02	3.71E+01	4.23E-01	4.00E-04	5.00E-04	8.46E-08
U-236	3.29E-02	3.71E+01	1.22E+00	4.00E-04	5.00E-04	2.44E-07
U-237	2.59E-01	3.71E+01	9.61E+00	4.00E-04	5.00E-04	1.92E-06
U-238	8.42E-02	3.71E+01	3.12E+00	4.00E-04	5.00E-04	6.25E-07
Np-237	8.81E-03	3.71E+01	3.27E-01	4.00E-04	5.00E-04	6.54E-08
Np-239	9.62E+00	3.71E+01	3.57E+02	4.00E-04	5.00E-04	7.14E-05
Pu-236	1.12E+02	3.71E+01	4.16E+03	4.00E-04	5.00E-04	8.31E-04
Pu-238	5.19E+01	3.71E+01	1.93E+03	4.00E-04	5.00E-04	3.85E-04
Pu-239	5.80E+01	3.71E+01	2.15E+03	4.00E-04	5.00E-04	4.30E-04
Pu-240	9.78E+03	3.71E+01	3.63E+05	4.00E-04	5.00E-04	7.26E-02
Pu-241	1.06E+04	3.71E+01	3.93E+05	4.00E-04	5.00E-04	7.87E-02
Am-241	5.17E+01	3.71E+01	1.92E+03	6.00E-04	5.00E-04	5.75E-04
Am-242m	3.41E-01	3.71E+01	1.27E+01	6.00E-04	5.00E-04	3.80E-06
Am-242	3.40E-01	3.71E+01	1.26E+01	6.00E-04	5.00E-04	3.78E-06
Am-243	9.62E+00	3.71E+01	3.57E+02	6.00E-04	5.00E-04	1.07E-04
Cm-242	4.90E+02	3.71E+01	1.82E+04	6.00E-04	5.00E-04	5.45E-03
Cm-243	4.90E+00	3.71E+01	1.82E+02	6.00E-04	5.00E-04	5.45E-05
Cm-244	2.75E+03	3.71E+01	1.02E+05	6.00E-04	5.00E-04	3.06E-02
Cm-246	2.15E-01	3.71E+01	7.98E+00	6.00E-04	5.00E-04	2.39E-06
	2.31E+05					2.50E+04

Table 3.4-8 Nuclides in RFA Gap at Time of Fuel Handling Accident

Nuclide	MAR, one RFA (Ci)	MAR, 2,800 RFAs (Ci)
H-3	5.16E+01	1.44E+05
Kr-85	1.05E+03	2.94E+06
Rn-219 (Noble Gas)	1.66E-02	4.65E+01
Rn-220 (Noble Gas)	2.36E+01	6.61E+04

Table 3.4-9 Fuel Breach Accident Source Term

Nuclide	MAR, one RFA (Ci)	MAR, 2800 RFAs (Ci)	DR	ARF	LPF	Source Term (Ci)
H-3	5.16E+01	1.44E+05	0.5	1.0	1.0	7.22E+04
Kr-85	1.05E+03	2.94E+06	0.5	1.0	1.0	1.47E+06
Rn-219	1.66E-02	4.65E+01	0.5	1.0	1.0	2.32E+01
Rn-220	2.36E+01	6.61E+04	0.5	1.0	1.0	3.30E+04

Table 3.4-10 Source Term for Seismic-Induced Breach of Waste Tank EP38 Over 7 Days

Isotope	MAR (Ci)	Damage Ratio (DR)	Airborne Release Fraction ^a	Source Term (Ci)
Cr-51	17.29	1.0	6.72E-04	1.2E-02
Co-60	0.32	1.0	6.72E-04	2.2E-04
Zn-65	0.85	1.0	6.72E-04	5.7E-04
Zr-95	1.38	1.0	6.72E-04	8.9E-04
Nb-95	0.74	1.0	6.72E-04	5.0E-04
Ru-106	8.90	1.0	6.72E-04	6.0E-03
Cs-134	0.74	1.0	6.72E-04	5.0E-04
Cs-137	1,018.66	1.0	6.72E-04	6.8E-01
Ce-144	10.18	1.0	6.72E-04	6.8E-03
Pu-240	1.06	1.0	6.72E-04	7.1E-04

a ARF here is really $ARR \times RFX \times 168$ hr.

Table 3.4-11 Sampled Isotope Fractions of the Resin Bed

Isotope	Curies	MAR
Cr-51	0.0179	0.0163
Co-60	0.0003	0.0003
Zn-65	0.0009	0.0008
Zr-95	0.0014	0.0013
Nb-95	0.0008	0.0007
Ru-106	0.0092	0.0084
Cs-134	0.0008	0.0007
Cs-137	1.0560	0.9610
Ce-144	0.0105	0.0096
Pu-240	0.0011	0.0010

Table 3.4-12 Resin Explosion Source Term Summary

Isotope	Fraction	MAR (Ci)	DRxARF	Source Term, (Ci)
Cr-51	0.0163	17.29	1E-03	1.7E-02
Co-60	0.0003	0.32	1E-03	3.2E-04
Zn-65	0.0008	0.85	1E-03	8.5E-05
Zr-95	0.0013	1.38	1E-03	1.4E-03
Nb-95	0.0007	0.74	1E-03	7.4E-04
Ru-106	0.0084	8.90	1E-03	8.9E-03
Cs-134	0.0007	0.74	1E-03	7.4E-04
Cs-137	0.9610	1,018.66	1E-03	1.0E+00
Ce-144	0.0096	10.18	1E-03	1.0E-02
Pu-240	0.0010	1.06	1E-03	1.1E-03
TOTAL		1,060		1.06

Table 3.4-13 Portable Deionizer Resin Fire Source Term

Isotope	MAR (Ci)	Airborne Release Fraction ^a	Source Term (Ci)
H-3	11.0	1.0E+00	1.1E+01
Cs-137	34.5	1.0E-02	3.5E-01
Cs-134	0.5	1.0E-02	5.0E-03
Sr-90	4.0	1.0E-02	4.0E-02
Nb-95	0.5	1.0E-02	5.0E-03

a From DOE-HDBK-3010-94 (Ref. 75)

Table 3.4-14 RBOF Radiological Consequences and Frequencies

Accident	OEP @ 100m (rem)	OEP @ 640m (rem)	MOI (rem)	Frequency (events/yr)
Process-Induced Criticality ^b	4.9E+00	1.60E-01	3.30E-03	6.64E-04
Fuel Rupture	5.30E-01	2.50E-02	3.30E-03	7.23E-01
NPH (earthquake)				
• criticality ^b	3.40E+00	1.10E-01	2.20E-03	2.55E-08
• fuel breach	5.30E+00	2.50E-01	3.30E-02	2.83E-08
• waste tank breach	2.05E-01	6.53E-03	8.10E-04	3.63E-03
Total NPH (earthquake)	2.05E-01 ^a	6.53E-03 ^a	8.10E-04 ^a	
NPH (high winds)				
• criticality ^b	3.40E+00	1.10E-01	2.20E-03	4.01E-05
• fuel breach	5.30E+00	2.50E-01	3.30E-02	3.93E-04
Total NPH (high winds)	8.70E+00	3.60E-01	3.52E-02	
Resin fire	3.40E-02	1.10E-03	1.40E-04	8.15E-04
Resin explosion	3.16E-01	1.01E-02	1.25E-03	2.50E-03
Uncontrolled chemical reaction	5.30E+00	2.50E-01	3.30E-02	2.16E-03

a Radiological consequences do not include events with frequency of less than 1E-06/yr.

b Radiological consequences provided by application of the 95% confidence methodology.

3.7 FIGURES

Figure 3-4.1 Criticality Yields for Metal & Oxide Systems

Figure 3.4-1 Criticality Yields for Metal & Oxide Systems

- 1 Process-Induced Criticality
- 2 Fuel Rupture
- 3 NPH (earthquake) Waste Tank Breach
- 4 NPH (high winds) Criticality
- 5 NPH (high winds) Fuel Breach
- 6 Resin Fire
- 7 Resin Explosion
- 8 Uncontrolled Chemical Reaction

Figure 3.4-2 Comparison to Guidelines for RBOF Accidents

Evaluation Basis Straight Wind Initiator	Structure Does Not Collapse	Struct. Failure Does Not Result in Crit.	CLASS	USER 1
IE-SWIND	NO-STRUCT	NO-CRIT-STRUCT		
			OK	SEQ-1W
			OK	SEQ-2W
			CRIT	SEQ-3W

Figure 3.4-3 Straight Wind Event Tree

Evaluation Basis Tornado Initiator	Structure Does Not Collapse	Struct. Failure Does Not Result in Crit.	CLASS	USER 1
IE-TORNADO	NO-STRUCT	NO-CRIT-STRUCT		
			OK	SEQ-1T
			OK	SEQ-2T
			CRIT	SEQ-3T

Figure 3.4-4 Tornado Event Tree

Evaluation Basis Seismic Initiator	Structure Does Not Collapse	Racks Do Not Fail	Failure(s) Does Not Result in Crit.	Basin Does Not Drain	CLASS	USER 1
IE-EQ	NO-STRUCT	NO-RACK	NO-CRIT	NO-DRAIN		
					OK	SEQ-1S
					DD	SEQ-2S
					OK	SEQ-3S
					DD	SEQ-4S
					CRIT	SEQ-5S
					CRIT/DD	SEQ-6S
					OK	SEQ-7S
					DD	SEQ-8S
					CRIT	SEQ-9S
					CRIT/DD	SEQ-10S
					OK	SEQ-11S
					DD	SEQ-12S
					CRIT	SEQ-13S
					CRIT/DD	SEQ-14S

Figure 3.4-5 Earthquake Event Tree

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 4

SAFETY STRUCTURES, SYSTEMS, AND COMPONENTS

AUGUST 1997

UNCLASSIFIED DOES NOT CONTAIN UNCLASSIFIED CONTROLLED NUCLEAR INFORMATION	
ADC & Reviewing Official:	<u>J.N. Dewes</u>
J.N. DEWES, Manager, Regulatory Programs	
Date:	<u>9/23/97</u>

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

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SAVANNAH RIVER SITE

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ACRONYMS

DOE	Department of Energy
LCO	Limiting Condition for Operation
NIM	Nuclear Incident Monitor
NPH	Natural Phenomena Hazard
PC	Performance Category
RBOF	Receiving Basin for Offsite Fuels
RRF	Resin Regeneration Facility
RRFC	Research Reactor Fuel Counter
SAR	Safety Analysis Report
SSCs	Structures, Systems, and Components
TSR	Technical Safety Requirement
WSRC	Westinghouse Savannah River Company

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4.0 SAFETY STRUCTURES, SYSTEMS, AND COMPONENTS

4.1 INTRODUCTION

This chapter of the Safety Analysis Report (SAR) for the Receiving Basin for Offsite Fuels (RBOF) and Resin Regeneration Facility (RRF) provides details on those facility Structures, Systems, and Components (SSCs) that are designated as safety class and safety significant.

The safety-class and safety-significant SSCs of the RBOF and RRF are necessary for the facility to satisfy the evaluation guidelines, provide defense in depth, or contribute to worker safety. The scope of this chapter includes the following:

- Description of the safety-class and safety-significant SSCs for the RBOF and RRF, including the required safety functions
- Identification of support systems upon which the safety-class and safety-significant SSCs depend to carry out the required safety functions
- Identification of the functional requirements necessary for the safety-class and safety-significant SSCs to perform their safety functions, and the general conditions caused by postulated accidents under which the safety-class and safety-significant SSCs must operate
- Identification of the performance criteria necessary to provide reasonable assurance that the functional requirements will be met
- Identification of all assumptions that require Technical Safety Requirement (TSR) coverage

As part of the safety analysis procedure defined in Department of Energy (DOE) Order 5480.23, the engineered safety features that prevent or mitigate the consequences of all potential accidents, including the design basis accidents, must be identified (Ref. 1). A description of the safety functions, interfaces, operating conditions, and performance criteria for the engineered safety features is also required.

DOE Order 5480.23 prescribes the use of a graded approach for the effort expended in safety analyses and the level of detail presented in associated documentation. The graded approach adjusts the magnitude of the preparation effort to the characteristics of the RBOF and RRF based on the facility hazard magnitude or severity, the facility complexity, and the facility life-cycle stage. Thus, DOE Order 5480.23 provides for developing the RBOF/RRF safety analysis based on these three factors.

The graded approach is based upon the hazard category of the facility under review. Hazard Category 2 and 3 facilities do not have the consequence potential associated with high hazard facilities, such as Class A reactors. Hazard Category 3 facilities will not have safety-class SSCs,

and the number of safety-significant SSCs, if any, will be less than that of a Hazard Category 2 facility due to the reduced magnitude of the hazards. The graded approach requires more formality in establishing functional requirements and performance criteria for safety-class SSCs than for safety-significant SSCs, since grading is a function of the potential hazard to the public.

The RBOF/RRF is a Hazard Category 2 facility and has safety-class functions (Ref. 2, 3).

4.2 REQUIREMENTS

This chapter was prepared in accordance with the format and content requirements defined in DOE Order 5480.23 and DOE-STD-3009-94 (Ref. 1, 4). Safety-class and safety-significant SSCs were identified in accordance with the guidance provided in DOE-STD-3009-94, Procedure 2.25 of Westinghouse Savannah River Company (WSRC) Procedure Manual E7, and WSRC-TM-93-9 (Ref. 5, 6). WSRC-TR-96-0093 (Ref. 3) documents the functional classification process for the RBOF and RRF.

Design codes and standards applicable to the RBOF and RRF are discussed in Chapter 2.

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4.3 SAFETY-CLASS STRUCTURES, SYSTEMS, AND COMPONENTS

The SSCs listed in Table 4.3-1 are designated as safety class. The table identifies the Chapter 3 accident(s) or rationale upon which the safety-class designation is based. The table also identifies the safety functions, functional requirements, and performance criteria judged to require TSR coverage.

Two additional safety-class functions for the RBOF and RRF are administratively controlled. To maintain both the radiological and chemical inventories as analyzed in Chapter 3, the quantity and physical form of the materials must be controlled. The TSR administrative programs that maintain these controls are the Fuel Receipt and Shipping Program and the Hazardous Material Inventory Control Program. Chapter 5 further discusses these programs. Also, to ensure that the basin water level is retained, operations procedures for monitoring and maintenance of basin water level provide both a safety-class and a safety-significant function and are classified as both.

4.3.1 FUEL BASIN AND CANAL SYSTEM

4.3.1.1 Safety Function

The safety function of the fuel basin and canal system is to contain water. The water contained within the basins serves as secondary or tertiary containment for fission products contained in the stored fuel. Also, the analysis in Chapter 3 assumes that during a high wind-induced criticality, the water provides a median value decontamination factor of 1,000 for halogens and 14,000 for particulate releases.. For process-induced criticality events, the basin water provides a median value decontamination factor of 1,000 for halogens and 2,500 for particulate releases. For either scenario, no credit is applied for decontamination of releases involving noble gases. The above decontamination factors limit the consequences to the offsite public. A basin water level of -19 inches with respect to floor grade provides adequate water depth to ensure that the decontamination factors used in the analysis are protected.

4.3.1.2 System Description

A description of the fuel basin and canal system is provided in Chapter 2.

The following drawings are referenced by the analysis of the basin structure (Ref. 7):

Drawing W232807, Rev. 46

Drawing W232808, Rev. 37

Drawing W232828, Rev. 25

Drawing W232829, Rev. 52

Drawing W232830, Rev. 59
Drawing W232831, Rev. 56
Drawing W233003, Rev. 31
Drawing W233004, Rev. 25
Drawing W233005, Rev. 62
Drawing W233075, Rev. 72
Drawing W233076, Rev. 58
Drawing W233077, Rev. 35
Drawing W233078, Rev. 26
Drawing W233079, Rev. 14
Drawing W233080, Rev. 6
Drawing W233083, Rev. 33
Drawing W233093, Rev. 22
Drawing W233094, Rev. 45
Drawing W233459, Rev. 49
Drawing W233485, Rev. 27
Drawing W233499, Rev. 2

4.3.1.3 Functional Requirements

The fuel basin and canal system must maintain structural integrity to the extent that it will prevent the loss of water contained in the basins below the level of -19 inches with respect to floor grade. The fuel basins must perform this function (1) before, during, and after an evaluation basis earthquake, straight winds, and tornadoes and (2) before and during an event which is initiated by the dropping of a shipping cask or heavy load. The fuel basin and canal system shall perform its function under the loading imposed by the collapse of the superstructure. The -19-inch water level is a pre-Natural Phenomena Hazard (NPH) event and pre-heavy load drop value maintained by the use of the basin water level indicator, which is discussed in Section 4.3.2. For the post-NPH event scenario, the basin water level indicator is not required since the basin has survived the event. Therefore, the water that was maintained prior to the NPH event is still present, providing water coverage for the fuel.

The -19-inch water level is a pre-heavy load drop value maintained by the use of the basin water level indicator, which is discussed in Section 4.3.2. The -19-inch water level is sufficient to account for the displaced basin water as a result of dropping the heavy load.

4.3.1.4 System Evaluation

A fragility study of the RBOF concrete structure, described in Chapter 2, shows that the basin structure is capable of withstanding loads imposed on it by a Performance Category (PC)-3 earthquake without significant cracking and leakage. Chapter 2 also concludes that the loads associated with complete failure of the Building 244-H superstructure will not cause major damage (cracking) to the concrete basins. Therefore, the basins will maintain their ability to retain the water required by the Chapter 3 analysis (Ref. 7). Additional details are provided in Chapters 2 and 3.

To address the concerns of basin leakage following a cask drop scenario, basin seepage analysis was performed to determine the rate at which the cask or storage basins would drain with various assumed crack sizes. The RBOF Basin Seepage Analysis (K-CLC-H-00110, Ref. 25) determined the time required to lower the basin water level such that the top of the tallest fuel assemblies would be exposed or a basin draindown of 4-1/2 feet below the full basin level. The basin cracks were assumed to be of constant width and either run longitudinally or transversely along the bottom of the basin and up its sides. From this analysis, the time required to lower the water level was determined to be approximately 400 hours for a crack width of 48 inches and 648 hours for a crack width of 2-1/2 inches. This would allow sufficient time for an integrated facility response to a basin leak.

4.3.1.5 Technical Safety Requirement Controls

The basin structure is considered a design feature, which is discussed in Chapter 2, and as such is not required to be specifically controlled by an Limiting Condition for Operation (LCO) in the TSR (Ref. 8). However, the Configuration Control Program covered in the administrative controls portion of the TSR along with the Design Features section of the TSR ensure that no unreviewed changes occur to safety-related design features. (Basin water level is controlled by the TSR; see Section 4.3.2.5.). The program requires that any design changes be reviewed to ensure that the basins can continue to withstand the effects of earthquakes, straight winds, tornadoes and the dropping of heavy loads in the basin and canal system.

4.3.2 BASIN WATER LEVEL INDICATOR

4.3.2.1 Safety Function

The basin water level indicator provides a visual indication of water level in the basins to ensure that the minimum level is present to provide the decontamination factor assumed in the analysis for criticality events.

4.3.2.2 System Description

The basin water level indicator is the sole SSC relied on to ensure the safety-class function of the fuel basins. The basin water level indicator is a passive measurement device (i.e., ruler) located in the #1 storage basin. Measurements are provided in inches with respect to floor grade. The ruler design included human factors considerations (Ref. 9).

4.3.2.3 Functional Requirements

The ruler is required to be intact and properly positioned to give an indication of water level to aid the operators in maintaining a level of at least -19 inches with respect to floor grade. The basin water level indicator is required prior to the occurrence of the accident. It is also required after a dropped cask (in cask basin with no fuel disruption) to enable operations to monitor water level and maintain the level. A dropped cask results in a potential crack and leak of basin water. It shall be of a material suitable for the normal operating environment. Markings shall be distinct and adequate to support the required accuracy of +/- 1/2 inch. The ruler does not require qualification for NPH events

4.3.2.4 System Evaluation

The safety function of this passive system is ensured by TSR controls. The location of the basin water level indicator precludes damage from a dropped shipping cask. Analysis from Chapter 3 indicates that monitoring every 112 hours during shutdown mode is adequate to ensure that no significant difference from -19 inches with respect to floor grade occurs during the unmonitored time period before an accident is initiated (Ref. 10).

4.3.2.5 Technical Safety Requirement Controls

The basin water level is maintained and controlled by an LCO which provides the RBOF basin water level limits of greater than -18 inches with respect to RBOF floor grade. The level is monitored at least every 12 hours during the Operations or Standby modes and every 112 hours during the Shutdown mode. If the level is not higher than -18 inches, the facility shall be placed in Standby or Shutdown Mode and corrective actions taken. Handling of fissile material is not permitted in the Standby or Shutdown modes.

4.4 SAFETY-SIGNIFICANT STRUCTURES, SYSTEMS, AND COMPONENTS

The SSCs listed in Table 4.4-1 are designated as safety significant. The table identifies the Chapter 3 accident(s) or rationale upon which the safety-significant designation is based. The table also identifies the safety functions, functional requirements, and performance criteria judged to require TSR coverage.

Some of the safety-significant functions in the RBOF and RRF are administratively controlled. Chapter 5 further discusses these programs and procedures and identifies specific elements that are credited in the hazards and accident analysis.

4.4.1 STORAGE RACKS AND ASSOCIATED HARDWARE

4.4.1.1 Safety Function

The storage racks, including associated hardware as defined in Chapter 2, serve to maintain fissile materials in a subcritical configuration as defined by the Nuclear Criticality Safety Program and to maintain spacing of the fuel assemblies to prevent fuel melting during a reduced water level condition.

4.4.1.2 System Description

Chapter 2 provides the system description of the storage racks.

4.4.1.3 Functional Requirements

The storage racks are required to function (maintain across-row and along-row spacing), during normal operations, scenarios involving a reduced water level, seismic events, and all process induced criticality events (excluding heavy load drop scenarios).

4.4.1.4 System Evaluation

The storage racks and associated hardware meet the safety function by providing adequate criticality spacing of fissile material when fuel is placed in proper storage locations. Proper storage locations are determined by Nuclear Safety Data Sheets. Due to the inherent flexibility of the fuel rack design, rack ties may become disengaged during fuel handling activities. An analysis of the impact of disengaged rack ties is presented in Chapter 2. It is concluded that disengagement or failure of rack ties does not result in a failure of the fuel racks to support fuel.

The storage racks and associated hardware also meet the safety function of providing adequate spacing of fuel to prevent fuel melting (surface area for heat dissipation) during conditions in

which a significant basin water reduction has occurred. Decay heat load for each fuel receipt is compared to the equivalent heat load as required by the Fuel Receipt and Shipping Program to ensure the fuel assemblies would not melt under the accident conditions postulated in Chapter 3.

A structural assessment of the racks in the #1 and #2 storage basins, described in Chapter 2, shows that the racks are capable of withstanding loads imposed on them by a PC-3 earthquake, provided certain fuel weight requirements are met (Ref. 11, 12). During evaluation basis wind and tornado events, the winds do not impact the storage racks directly, but it is possible for the winds to damage the RBOF building structure and for the debris to fall into the storage basins and impact the racks. The racks provide some protection against fuel breach and criticality as configured from this falling debris. However, the racks are not assumed to prevent wind-induced debris from falling into the basin and causing a fuel breach or criticality as discussed in Chapter 3. Additional details on the storage racks and associated hardware are provided in Chapter 2.

4.4.1.5 Technical Safety Requirement Controls

The storage racks are considered a design feature, which is discussed in Chapter 2, and as such are not required to be specifically controlled by an LCO in the TSR (Ref. 8). However, the Configuration Control Program covered in the administrative controls portion of the TSR along with the Design Features section of the TSR ensure that no unreviewed changes occur to safety-related design features. This program requires that any design changes be reviewed to ensure that the structural integrity of the racks or hardware will not be affected by the proposed activity and that the storage racks can continue to withstand the effects of earthquakes, reduce the probability of a criticality due to tornadoes and straight winds, and maintain adequate fuel spacing.

4.4.2 FUEL BASIN AND CANAL SYSTEM

4.4.2.1 Safety Function

The safety function of the fuel basin and canal system is to maintain system integrity and contain water. The basin piping penetration design is such that a complete basin draindown due to Filter Deionizer system operation is not postulated. The walls of the basins and the water contained in the basins provide shielding to the worker during normal operations and criticality accident conditions, and serve as secondary or tertiary containment for fission products in the fuel contained in the basin water as discussed in Section 4.3.1.1. Because of the shielding provided, the worker has sufficient time to evacuate to a safe area.

4.4.2.2 System Description

The system description for the fuel basin and canal system is provided in Chapter 2.

4.4.2.3 Functional Requirements

Fuel handling and/or storage basins and associated transfer canals are required to maintain 4-1/2 feet of basin water above the fuel, for shielding, during conditions of normal operations and criticality events, including NPH events. For conditions of normal operation, this requirement can be waived if constant RCO or radiation monitoring is provided. A basin level of -19 inches with respect to floor grade corresponds to a minimum depth of 4-1/2 feet above the fuel. For process error criticality events, this level will provide adequate shielding and containment of fission products to protect the worker provided the worker is able to evacuate the affected area (Ref. 13). In addition, the basin walls provide shielding to workers outside the RBOF building (Ref. 14).

4.4.2.4 System Evaluation

Section 4.3.1.4 provides the system evaluation of the fuel basin and canal system. Also, the Chapter 3 analysis of process criticality demonstrates that the water will provide adequate containment of fission products and shielding of the facility worker from a process error criticality event. However, for the high wind-induced criticality events, the shielding and containment may be inadequate because of the possible inability of the facility worker to evacuate to a safe area. The fuel basin and canal system is qualified to a PC-3 earthquake. N-NCS-H-00011 (Ref. 14) demonstrated that the basin walls provide adequate shielding for workers outside the RBOF building.

4.4.2.5 Technical Safety Requirement Controls

Section 4.3.1.5 provides the TSR controls for the fuel basin and canal system.

4.4.3 BASIN WATER LEVEL INDICATOR

4.4.3.1 Safety Function

The safety-significant function of this SSC, which is to indicate basin water level, is provided for worker safety. This SSC, along with visual monitoring by the operators, ensures that an adequate level of water is present in the basin to provide shielding (at least -19 inches with respect to floor grade) and to provide secondary or tertiary containment for radionuclides contained in the stored fuel.

4.4.3.2 System Description

Section 4.3.2.2 provides the system description for the basin water level indicator.

4.4.3.3 Functional Requirements

Section 4.3.2.3 provides the functional requirements for the basin water level indicator.

4.4.3.4 System Evaluation

Section 4.3.2.4 provides the system evaluation for the basin water level indicator.

4.4.3.5 Technical Safety Requirement Controls

Section 4.3.2.5 provides the TSR controls for the basin water level indicator.

4.4.4 SHIELDING

4.4.4.1 Safety Function

The facility design provides shielding at various locations to protect personnel from direct gamma radiation.

4.4.4.2 System Description

Reinforced-concrete walls surround the waste cell, decontamination cell, and resin regeneration cell. Personnel within the RBOF control room are shielded from radiation emanating from the basins by a concrete shield wall. A shield wall in the RRF control room shields personnel from radiation emanating from the resin regeneration cell and from the portable deionizers positioned adjacent to the hose box station (Ref. 15).

4.4.4.3 Functional Requirements

The shielded cells are required to function (i.e., provide attenuation of radiation) during release of radioactive material to the cells, for accidents not induced by NPHs. The control room shield walls are required to function during events involving the release of radioactivity from the basins, the resin regeneration cell, and the portable deionizers.

4.4.4.4 System Evaluation

The attenuation of radiation that is provided by the shielded cells and shield walls under normal operating conditions has been confirmed by the Radiation Protection Program discussed in WSRC Procedure Manual 5Q (Ref. 16). Adequacy under accident conditions has been qualitatively determined (Ref. 17).

4.4.4.5 Technical Safety Requirement Controls

The shielded cells and shield walls are considered a design feature, which is discussed in Chapter 2, and as such are not required to be specifically controlled by an LCO in the TSR (Ref. 8). However, the Configuration Control Program covered in the administrative controls portion of the TSR along with the Design Features section of the TSR ensure that no unreviewed changes occur to safety-related design features. The program requires that any design changes be reviewed to ensure that the shielding can continue to perform its safety-significant function.

4.4.5 ONSITE CASKS

4.4.5.1 Safety Function

Onsite casks have safety-significant functions for worker protection. These functions include shielding workers from direct radiation, preventing the release of radioactive materials contained in the cask, and preventing an inadvertent criticality (Ref. 18).

4.4.5.2 System Description

Section 2.6.1.1 describes the onsite casks.

4.4.5.3 Functional Requirements

The safety-significant functional requirements of the onsite casks are confinement of radioactive material, shielding from direct radiation, and maintenance of a subcritical configuration. These functional requirements do the following:

- Provide adequate confinement to ensure that no worker receives a dose that would cause prompt death or permanent injury
- Prevent any individual from receiving a dose of 25 rem or greater from direct radiation or from the release of radioactive material from the cask
- Prevent a criticality involving the material in the cask

The cask must be resistant to leaks as discussed in Section 3.3.

4.4.5.4 System Evaluation

Onsite casks are authorized for use in accordance with site procedures (Ref. 9). The analysis of accidents involving the onsite casks conclude that the onsite consequences will be low.

Section 3.3 provides additional details.

4.4.5.5 Technical Safety Requirement Controls

The cask configuration (e.g., insert used) and cadmium plate operability are controlled by an LCO which ensure the cask is configured for the fuel to be shipped in accordance with an approved NSDS. If the cask is not configured properly, or the cadmium plates are found to not provide the required reduction in neutron transmission, the cask is removed from service, or corrective actions are taken. The SRS 70 Ton Cask is also considered a design feature. The Configuration Control Program covered in the administrative controls portion of the TSR along with the Design Features section of the TSR ensure that no unreviewed changes occur to safety-related design features. This program requires that any design changes be reviewed to ensure that the integrity of the cask will not be affected by the proposed activity. The Fuel Receipt and Shipping Program will ensure that during transportation the safety analysis is addressed in accordance with WSRC Procedure Manual 19Q (Ref. 19). The Fuel Receipt and Shipping Program is further discussed in Chapter 5.

4.4.6 RRFC SKIRT

4.4.6.1 Safety Function

The skirt plates which are located at the bottom of the Research Reactor Fuel Counter (RRFC) prevent any dropped assembly from inadvertently rolling under the RRFC while another assembly is positioned in the RRFC. This configuration could potentially cause a criticality event.

4.4.6.2 System Description

The mechanical skirt consists of four 7 inch by 21 inch by 1/8 inch stainless steel plates bolted onto the RRFC's bottom plate (Ref. 20).

4.4.6.3 Functional Requirements

The RRFC skirt plates isolate the perimeter of the space underneath the RRFC bottom plate to prevent any dropped fuel assembly with a rectangular or oval cross-section greater than 2 inches in width or diameter from inadvertently being positioned below the assay chamber.

4.4.6.4 System Evaluation

The fabrication and installation of the RRFC skirt plates have been inspected to ensure the as-built configuration prevents any dropped assembly from inadvertently rolling under the RRFC while another assembly is positioned in the RRFC.

4.4.6.5 Technical Safety Requirement Controls

The RRFC skirt plates are considered a design feature, which is discussed in Chapter 2, and as such are not required to be specifically controlled by an LCO in the TSR (Ref. 8). However, the Configuration Control Program covered in the administrative controls portion of the TSR along with the Design Features section of the TSR ensure that no unreviewed changes occur to safety-related design features.

4.4.7 NUCLEAR INCIDENT MONITOR SYSTEM

The Nuclear Incident Monitor (NIM) system is a safety-significant SSC. The evacuation of the area is also a safety-significant function, which is controlled by the Emergency Preparedness Program.

4.4.7.1 Safety Function

The NIM system has the safety-significant function of monitoring, detecting, and providing an alarm upon a criticality with a fission product release or upon other accidents (e.g., fuel exposure, fuel rupture, or high radiation area) involving direct radiation dose from shine or inhalation of fission product gas releases.

4.4.7.2 System Description

In general, NIM systems are provided wherever it is deemed that they will result in a reduction in total risk to the facility workers. The NIM system is capable of detecting excessive amounts or intensities of radiation, which may be due to a criticality or other accidents (e.g., fuel exposure, fuel rupture, or high radiation area), and of signaling immediate personnel evacuation. NIM instruments continually monitor each potential criticality site, except as provided in WSRC-SCD-3 (Ref. 21).

The RBOF NIM system uses NIM pairs. When used in pairs, each NIM instrument alarms individually if the radiation setpoint is reached. The NIM pairs require only one of the NIM instruments to actuate the auxiliary alarm system. The NIM system provides coverage for monitoring, detecting, and alarming when fission products are released (except as provided in WSRC-SCD-3). This system includes a criticality detection and alarm function.

Chapter 2 provides a more detailed discussion of the NIM system, and Chapter 6 provides a discussion of the number, location, and testing of the NIM systems used at the RBOF.

4.4.7.3 Functional Requirements

The NIM system is required to provide coverage and to alarm when it detects fission products released as a result of a criticality or other accidents involving direct radiation dose or fission product gas release.

4.4.7.4 System Evaluation

According to WSRC-SCD-3 (Ref. 21), testing of the alarm system response and performance may be accomplished without requiring evacuation. A NIM unit is returned to operating condition immediately following tests. NIM system response to radiation is measured to confirm continuing instrument performance, and the NIM system is recalibrated periodically. Records of tests and recalibration are maintained. The NIM alarm system is tested periodically. Field tests verify that the signal is audible above background noise. Alarm decibel levels are also measured following an original baseline measurement. The NIM bell system, part of the NIM alarm system, requires power and is lost during a power outage. The NIM unit and its alarm are powered by backup power from the H-Area tank farm and are backed up by battery power for 24 hours after a power outage.

Initial testing and inspection of the NIM system verified that the system was fabricated and installed in accordance with plans and specification. If the system is modified or repaired, the system is tested and inspected in a manner equivalent to the initial installation tests and inspections. The design and installation are resistant to earthquake damage. The system is designed for high reliability and uses components that do not require frequent servicing.

The NIM system is controlled by a LCO in the TSR.

4.4.7.5 Technical Safety Requirement Controls

Detailed TSR requirements ensure the operability of the NIMs. Evacuation of the area is controlled by the Emergency Preparedness Program.

4.4.8 CRANE LIMIT SWITCH SYSTEM

4.4.8.1 Safety Function

The crane limit switch system stops the crane movement, preventing the crane from traveling over the #2 storage basin and contacting the mechanical stops, thus preventing the 100-ton crane from potentially dropping a load into the basin. If the crane did contact the mechanical stops when loaded, the possibility of dropping a load into the #2 storage basin would exist.

4.4.8.2 System Description

The crane limit switch system prevents the 100-ton crane from contacting the mechanical stops. When actuated by a stop bar, the limit switch provides braking capability to the crane motor. This limit switch is set to actuate before the crane can reach the mechanical stops. Additional details of the crane limit switch system are provided in Chapter 2.

4.4.8.3 Functional Requirements

The 100-ton crane south travel limit switch actuation bar is positioned to prevent the crane from traveling over the #2 storage basin and contacting the mechanical stops. If the 100-ton crane did contact the south end of the rail mechanical stops when loaded, the possibility of dropping the load into the #2 storage basin would exist. The limit switches are actuated by a crane stop bar mounted on the building structure to limit the crane travel by removing power to the crane motors. The location of the actuator bar is approximately 29 feet north of Column line 6. Chapter 2 specifies that this location prevents the crane from contacting the mechanical stops even with the crane's momentum from a maximum speed.

4.4.8.4 System Evaluation

Periodic procedural testing ensures that the stop bar will actuate the limit switch. Between tests, an instrument loop test is performed on the limit switch, and the position of the crane north of the #2 storage basin is verified during operation and standby (every 24 hours), within 4 hours prior to entering shutdown, and every 7 days during shutdown.

4.4.8.5 Technical Safety Requirement Controls

The limit switch is controlled by an LCO in the TSR. Detailed TSR requirements ensure the operability of the crane limit switch system as described above. The TSR requires suspension of all north-south motion of the crane if the limit switch is inoperable. If the switch cannot be restored to operable status or if the crane cannot be repositioned immediately, a response plan shall be initiated in accordance with the TSR within 24 hours.

4.4.9 BUILDING 244-H STRUCTURE

4.4.9.1 Safety Function

The Building 244-H structure will maintain its integrity to prevent a criticality or a fuel breach caused by its failure during a seismic event. The 100-Ton Crane rails and associated mechanical stops are considered an integral part of the 244-H structure. An additional safety function of the

mechanical stops is to prevent the 100-Ton crane from derailing at the ends of the rails, and potentially initiating a criticality event in the storage basins.

4.4.9.2 System Description

Building 244-H is described in Chapter 2.

4.4.9.3 Functional Requirements

The wind fragility analysis shows that Building 244-H adequately supports the performance goals assumed in Chapter 3. The building is qualified to withstand the PC-3 evaluation basis earthquake.

4.4.9.4 System Evaluation

Chapter 3 shows that Building 244-H is qualified to withstand a PC-3 evaluation basis earthquake. The safety analysis is supported by the fragility analysis for straight winds and tornadoes. Chapter 3 provides a discussion of the fragility analysis.

4.4.9.5 Technical Safety Requirement Controls

Building 244-H is considered a design feature, which is discussed in Chapter 2, and as such is not required to be specifically controlled by an LCO in the TSR (Ref. 8). However, the Configuration Control Program covered in the administrative controls portion of the TSR along with the Design Features section of the TSR ensure that no unreviewed changes occur to safety-related design features. The program requires that any design changes be reviewed to ensure that the building can continue to withstand a straight wind or earthquake as described in Chapter 3.

4.4.10 NITRIC ACID TANK

4.4.10.1 Safety Function

The nitric acid tank is required to prevent the interaction of nitric acid and sodium nitrite. The tank prevents nitric acid from being displaced to the Chemical pad No. 2 during PC-2 high winds or earthquake and potentially mixing with sodium nitrite.

4.4.10.2 System Description

The nitric acid tank is a portable dumpster. This 1,000-gallon horizontal tank (assumed volume of 1,100 gallons, including head space) containing 50% nitric acid is constructed of stainless

steel supported by continuous saddle. It rests on a 4-inch concrete pad. All of this equipment is supported by a 6-inch concrete slab on grade. The tank is located about 21 inches from a similar tank containing sodium hydroxide (Ref. 22).

4.4.10.3 Functional Requirements

The nitric acid tank prevents a spill which could potentially mix with any spilled sodium nitrite within the Chemical pad No. 2 bermed area during high wind and seismic conditions.

4.4.10.4 System Evaluation

The nitric acid tank will remain within the berm during all wind loads resulting from a PC-2 high wind event or a PC-2 seismic event (Ref. 22).

4.4.10.5 Technical Safety Requirement Controls

The nitric acid tank is considered a design feature, which is discussed in Chapter 2, and as such is not required to be specifically controlled by an LCO in the TSR (Ref. 8). However, the Configuration Control Program covered in the administrative controls portion of the TSR along with the Design Features section of the TSR ensure that no unreviewed changes occur to safety-related design features. The program requires that any design changes be reviewed to ensure that the tank can continue to perform its safety-significant function.

4.4.11 BERM AT CHEMICAL PAD NO. 2

4.4.11.1 Safety Function

The berm at chemical pad No. 2 prevents spilled sodium nitrite from overflowing into the drainage system and potentially mixing with spilled nitric acid.

4.4.11.2 System Description

The berm at chemical pad No. 2 is a concrete berm of at least 3 inches in height for the entire length. The berm completely surrounds chemical pad No. 2. The area is a rectangle with dimensions of 270 inches by 348 inches, with a 66-inch by 146-inch rectangular section removed from one corner, which gives a total surface area of 585.5 ft² (Ref. 23).

4.4.11.3 Functional Requirements

The berm at chemical pad No. 2 is required to ensure that all leaks remain inside the bermed area, including those caused by a high wind or seismic-induced rain shelter failure event, to prevent a sodium nitrite leak from the berm reacting with nitric acid and forming nitrous dioxide.

4.4.11.4 System Evaluation

The berm contains a volume of 963 gallons (Ref. 23). Chapter 3 states that the maximum volume of sodium nitrite permitted on the bermed area is 900 gallons.

Vehicular impact, seismic, and rain shelter failure effects on the berm have been calculated. It has been established that the resulting demand stress due to flexure and shear on the berm are well within the capacity of the berm. Therefore, the berm is expected to withstand these impacts without cracking. Section 3.3 provides additional details of the berm at chemical pad No. 2.

4.4.11.5 Technical Safety Requirement Controls

The berm at chemical pad No. 2 is considered a design feature, which is discussed in Chapter 2, and as such is not required to be specifically controlled by an LCO in the TSR (Ref. 8). However, the Configuration Control Program covered in the administrative controls portion of the TSR along with the Design Features section of the TSR ensure that no unreviewed changes occur to safety-related design features. The program requires that any design changes be reviewed to ensure that the berm shall continue to perform its safety-significant function.

4.4.12 SODIUM NITRITE TANK

4.4.12.1 Safety Function

The sodium nitrite tank is required to not be displaced off chemical pad No. 2 during PC-2 high wind and seismic events.

4.4.12.2 System Description

The sodium nitrite tank is a portable dumpster. This 1,000-gallon horizontal tank (assumed volume of 1,100 gallons, including head space) containing 30% sodium nitrite is constructed of stainless steel supported by continuous saddle. It rests on a 4-inch concrete pad. All of this equipment is supported by a 6-inch concrete slab on grade. Chapter 2 provides additional details of the sodium nitrite tank.

4.4.12.3 Functional Requirements

The tank prevents a sodium nitrite spill outside the berm under high wind or seismic conditions by ensuring that the tank remains on the berm during these events.

4.4.12.4 System Evaluation

Analysis has proven that the sodium nitrite tank will survive all wind loads resulting from PC-2 high wind and seismic events (Ref. 24).

4.4.12.5 Technical Safety Requirement Controls

The sodium nitrite tank is considered a design feature, which is discussed in Chapter 2, and as such is not required to be specifically controlled by an LCO in the TSR (Ref. 8). However, the Configuration Control Program covered in the administrative controls portion of the TSR along with the Design Features section of the TSR ensure that no unreviewed changes occur to safety-related design features. The program requires that any design changes be reviewed to ensure that the tank can continue to perform its safety-significant function.

4.4.13 SODIUM NITRITE HOLD TANK

4.4.13.1 Safety Function

The sodium nitrite hold tank is required to not be displaced off chemical pad No. 2 during PC-2 high wind and seismic events.

4.4.13.2 System Description

The sodium nitrite hold tank is described in Chapter 2.

4.4.13.3 Functional Requirements

The tank prevents a sodium nitrite spill outside the berm under high wind or seismic conditions by ensuring that the tank remains on the berm during these events.

4.4.13.4 System Evaluation

Analysis has proven that the sodium nitrite hold tank will survive all wind loads resulting from PC-2 high wind and seismic events (Ref. 24).

4.4.13.5 Technical Safety Requirement Controls

The sodium nitrite hold tank is considered a design feature, which is discussed in Chapter 2, and as such is not required to be specifically controlled by an LCO in the TSR (Ref. 8). However, the Configuration Control Program covered in the administrative controls portion of the TSR along with the Design Features section of the TSR ensure that no unreviewed changes occur to safety-related design features. The program requires that any design changes be reviewed to ensure that the tank can continue to perform its safety-significant function.

4.4.14 REMOTE ALARMS IN 221-H CONTROL ROOM

4.4.14.1 Safety Function

The Remote 244-H General Alarm provides indication of a NIM trouble alarm to the command function personnel in the Building 221-H Control room during a RBOF facility shutdown. The Remote 244-H General Alarm also provides a non-safety-related indication to the Building 221-H Control Room of a potential problem in RBOF during Shutdown.

The Remote NIM Alarm provides indication of high direct radiation to the command function personnel in the Building 221-H Control Room during a RBOF facility shutdown. The remote NIM Alarm also provides an indication to site workers of a potential high radiation area in RBOF during Shutdown.

4.4.14.2 System Description

The 244-H General Alarm is actuated by various RBOF operational alarms with the exception of the Continuous Air Monitors and Gamma Monitors. During the operation and standby modes, a key switch in the RBOF control room is positioned in the Local position to actuate the alarm in the RBOF control room. During shutdown, the RBOF/RRF command function is transferred to the 221-H Control Room. To facilitate this transfer, the 221H General Alarm Bypass key switch is positioned in the Remote position which will notify the 221-H control room of a potential problem in RBOF. The input of concern for the 244-H General Alarm is the NIM trouble alarm which actuates upon receiving a NIM trouble (malfunction or misoperation) signal. The other alarm inputs are not required to fulfill the safety significant function.

The Remote NIM Alarm is connected to the 3 pairs of NIM units in RBOF. Actuation of any of the six RBOF NIM units will result in both visual (panel light) and audible (alarm horn) indication in the 221-H control room. There is no transfer function associated with this remote alarm.

4.4.14.3 Functional Requirements

The 244-H General Alarm notifies the 221-H control room of a potential problem with the NIM units in RBOF. This alarm prevents personnel from entering RBOF during a postulated scenario where a concurrent reduction in water level (increased radiation level) and NIM unit failure exists.

The Remote NIM Alarm notifies the 221-H control room of a potential high radiation area in RBOF during Shutdown.

4.4.14.4 System Evaluation

The operability of the 244-H General Alarm and the Remote NIM Alarm are verified prior to entering shutdown. In addition, the Remote NIM Alarm is verified periodically during facility shutdown by ensuring the control room light is functioning. This verification does not involve a test of the individual RBOF NIM units or of the 221-H audible Nuclear Incident Alarm.

4.4.14.5 Technical Safety Requirement Controls

Detailed TSR requirements ensure the operability of the Remote 244-H General Alarm and NIM Alarm as described above. If inoperable, the TSR requires the facility to restore the alarms to operable status within 12 hours while in Shutdown. If the alarms cannot be restored to operable status, the facility is returned to Standby status within 8 hours or is required to develop a response plan in accordance with the TSR within 24 hours.

4.4.15 FUEL TRANSPORT AND STORAGE EQUIPMENT

4.4.15.1 Safety Function

The fuel transport and storage equipment, which includes: fuel bundles, buckets and bucket inserts, test tubes, cans, and blocking devices, but does not include the tools (commonly referred to as "Peters Tools, J-Hooks, etc.). The fuel transport and storage equipment maintain safe fuel configurations (mass and spacing) during all normal operating and accident conditions, and to reduce the frequency of process-induced criticality events.

4.4.15.2 System Description

During cask unloading processes, the transfer bucket, fitted with a specified insert, provides a level of control on the number of fuel elements as they are unloaded from the cask. The insert ensures that all subsequent handling of the fuel is safe (i.e., total amount of fuel being handled at any point in time is less than or equal to a Safe Fuel Unit as defined by the NSDS). Once loaded,

the transfer bucket is used to transfer the fuel elements to a working basin for further processing (cropping, bundling, etc.) or staging. Movement of the transfer bucket to the canal is accomplished by means of a traveling bridge.

Other buckets are also used for unloading, identification, isolation, and other miscellaneous fuel handling/storage activities that do not use inserts, and thus are covered by the administrative controls in place under the Criticality Safety Program. These buckets are typically used to handle only one fuel element or for fuel storage in isolation.

Tools are used to remove a fuel assembly and to place it in the unloading bucket. The unloading bucket is then moved to a position on the cask basin floor near the transfer bucket. Unloading and transfer buckets are separated by an isolation bucket. Tools are used again to remove the assembly from the unloading bucket and to place it on a bundling table for identification.

In some situations, cask baskets are removed directly from the cask (e.g., legal weight truck cask) and are placed in basket stands on the basin floor prior to fuel unloading. Prior to removing the loaded cask basket, the basin is cleared of unsecured fuel. The cask basket is lifted and placed on the basin floor, and fuel unit unloading is accomplished as described above. Administrative restrictions generally prohibit removal from casks of cask baskets loaded with more than the safe number of fuel pieces unless specific analyses have been performed to authorize such operations.

4.4.15.3 Functional Requirements

The fuel transport and storage equipment is used maintain fuel spacing and mass in safe configurations to prevent or reduce the frequency of criticality events. These SSCs are required to maintain integrity during normal operating and accident conditions.

4.4.15.4 System Evaluation

The Chapter 3 analysis of process-induced criticality events relies upon the passive spacing and mass controls provided by the fuel transport and storage equipment, in conjunction with the operating procedures, to prevent or reduce the frequency of a process-induced criticality event. The proper utilization of these passive devices ensures neutronic isolation between safe fuel units and groups. The blocking devices are used where applicable to ensure that fuel being handled cannot be dropped into an unsafe configuration.

4.4.15.5 Technical Safety Requirement Controls

The fuel transport and storage equipment is considered a design feature, which is discussed in Chapter 2, and as such are not required to be specifically controlled by an LCO in the TSR (Ref. 8). However, the Configuration Control Program covered in the administrative controls portion of the TSR along with the Design Features section of the TSR ensure that no unreviewed changes occur to safety-related design features. The program requires that any design changes be reviewed to ensure that the fuel transport and storage equipment can continue to perform its safety-significant function.

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4.6 TABLES

Table 4.3-1 Safety-Class Structures, Systems, and Components

Chapter 4 Section	Safety-Class SSCs	Chapter 3 Accident / Rationale	Safety Function	Functional Requirements	Performance Criteria
4.3.1	Fuel Basin and Canal System.	Earthquakes, straight winds, tornadoes, and criticality events.	Contain water.	Retain the ability to maintain a water level of -19 inches or higher with respect to floor grade during accident conditions.	Passive component. Design feature. Configuration control.
4.3.2	Basin Water Level Indicator.	Normal conditions.	Provide a visual indication of water level.	Provide water level indication for monitoring by operator.	Passive component. Monitoring controlled by TSR LCO to a basin water level of -18 inches or higher with respect to floor grade.

Table 4.4-1 Safety-Significant Structures, Systems, and Components

Chapter 4 Section	Safety-Significant SSCs	Chapter 3 Accident / Rationale	Safety Function	Functional Requirements	Performance Criteria
4.4.1	Storage Racks and Associated Hardware. Refer to Table 2.4-1.	Earthquakes, straight winds, tornadoes, criticality events, and normal operating conditions.	Maintain fissile materials in a subcritical spacing as determined by the Nuclear Criticality Safety Program.	Maintain across-row and along-row spacing.	Passive component. Configuration control. Spacing requirements are set by programs specified in the Nuclear Criticality Safety Program.
4.4.2	Fuel Basin and Canal System.	Earthquakes, straight winds, tornadoes, criticality events, and normal operating conditions.	Contain water.	Retain the ability to maintain a water level of -19 inches or higher with respect to floor grade.	Passive component. Design feature. Configuration control.
4.4.3	Basin Water Level Indicator.	Earthquakes, straight winds, tornadoes, criticality events, normal operating conditions and certain cask drop events.	Provide a visual indication of water level.	Provide water level indication for monitoring by operator.	Passive component. Monitoring controlled by TSR LCO to a basin water level of -18 inches or higher with respect to floor grade.

Table 4.4-1 Safety-Significant Structures, Systems, and Components (Continued)

Chapter 4 Section	Safety-Significant SSCs	Chapter 3 Accident/Rationale	Safety Function	Functional Requirements	Performance Criteria
4.4.4	Shielding.	Criticality and radiological releases, and normal operations.	Protect personnel from direct gamma radiation.	Attenuate radiation during accident conditions for shielded cells, and radiation releases from the basins and the portable deionizers.	Passive components. Design features. Configuration control.
4.4.5	Onsite Casks.	Releases from the cask, in-cask criticalities.	Provide shielding from direct radiation, prevent release of radioactive materials contained in the cask, and prevent inadvertent criticality.	Provide confinement such that no site worker receives a dose that would cause prompt death or permanent injury and no other individual receives a dose of 25 rem. Prevent inadvertent criticality.	Authorized for use by the WSRC Procedure Manual 19Q.
4.4.6	RRFC Skirt	Criticality event.	Prevent assembly being located under RRFC.	Prevents objects greater than 2 inches in width or diameter from falling under RRFC	Passive component. Design feature. Configuration control.
4.4.7	Nuclear Incident Monitor System.	During event involving potential radiation exposure.	Monitor and detect radiation.	Detection of radiation.	NIM operability is ensured by TSR requirements.

Table 4.4-1 Safety-Significant Structures, Systems, and Components (Continued)

Chapter 4 Section	Safety-Significant SSCs	Chapter 3 Accident/Rationale	Safety Function	Functional Requirements	Performance Criteria
4.4.8	Crane Limit Switch System.	Criticality or fuel rupture initiated by crane dropping load into fuel basin.	Prevent crane from dropping load into basin.	Crane stop bar actuates the switch, preventing movement of the crane to impact the mechanical stops.	Periodic testing that the stop bar actuates the switch. Instrument loop test on the switch.
4.4.9	Building 244-H Structure.	Earthquake, straight winds, tornadoes, and criticalities.	Function during an NPH event to support the Chapter 3 analysis. Shielding is covered in Section 4.4.4.	The wind fragility analysis must show that Building 244-H adequately supports the performance goals assumed in Chapter 3 of this SAR. Qualified to PC-3 earthquake.	Passive component. Design feature. Configuration control.
4.4.10	Nitric Acid Tank.	Nitric acid spills, earthquake, straight winds, and tornadoes.	Retain tank in the berm during high winds and earthquake.	Qualified to stay on berm for PC-2 wind and earthquake.	Passive component. Changes to design must be reviewed to ensure that the tank can continue to perform its function.
4.4.11	Berm at Chemical Pad No. 2.	Sodium nitrite spill caused by any credible mechanism.	Limit evaporative area. Prevent leaks outside the bermed area.	Qualified for wind failure of rain shelter, vehicle impact, and earthquake.	Passive component. Design feature. Configuration control.

Table 4.4-1 Safety-Significant Structures, Systems, and Components (Continued)

Chapter 4 Section	Safety- Significant SSCs	Chapter 3 Accident/ Rationale	Safety Function	Functional Requirements	Performance Criteria
4.4.12	Sodium Nitrite Tank.	Sodium nitrite spills, earthquake, straight winds, and tornadoes.	Retain tank in the berm during high winds and seismic event.	Qualified to stay on berm for PC-2 wind and earthquake.	Passive component. Changes to design must be reviewed to ensure that the tank can continue to perform its function.
4.4.13	Sodium Nitrite Hold Tank.	Sodium nitrite spills, earthquake, straight winds, and tornadoes.	Retain tank in the berm during high winds and seismic event.	Qualified to stay on berm for PC-2 wind and earthquake.	Passive component. Changes to design must be reviewed to ensure that the tank can continue to perform its function.
4.4.14	Remote Alarms	FD system leaks with or without NIM power failure	Prevent inadvertent entry into high radiation area	Provide remote indication of RBOF equipment failure	Remote alarms operability are ensured by TSR requirements
4.4.15	Fuel transport and storage equipment specified to maintain safe fuel configurations (fuel bundles, buckets and bucket inserts, test tubes, cans, and blocking devices).	Criticality events and normal operating conditions.	Maintain fuel configurations to prevent or reduce the frequency of criticality events.	These SSCs are required to maintain integrity during normal operating and accident conditions..	Passive component. Configuration control. Spacing requirements are set by programs specified in the Nuclear Criticality Safety Program

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 5

DERIVATION OF TECHNICAL SAFETY REQUIREMENTS

AUGUST 1997

UNCLASSIFIED DOES NOT CONTAIN UNCLASSIFIED CONTROLLED NUCLEAR INFORMATION	
ADC & Reviewing Official:	<u>J.N. Dewes</u>
J.N. DEWES, Manager, Regulatory Programs	
Date:	<u>9/23/97</u>

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

SAVANNAH RIVER SITE

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SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500.

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ACRONYMS

ALARA	As Low As Reasonably Achievable
ARM	Area Radiation Monitor
CFR	Code of Federal Regulations
DBA	Design Basis Accident
DOE	Department of Energy
EG	Evaluation Guideline
ERPG	Emergency Response Planning Guideline
GBA	General Building Area
ICR	Incremental Cancer Risk
LCO	Limiting Condition for Operation
LCS	Limiting Control Setting
NIM	Nuclear Incident Monitor
NPH	Natural Phenomena Hazard
NSDS	Nuclear Safety Data Sheet
RBOF	Receiving Basin for Offsite Fuels
RCO	Radiological Control Operations
RFA	Reference Fuel Assembly
RRF	Resin Regeneration Facility
RRFC	Research Reactor Fuel Counter
SAR	Safety Analysis Report
SL	Safety Limit
SR	Surveillance Requirement
SRS	Savannah River Site
SSC	Structure, System, and Component
TSR	Technical Safety Requirement
USQ	Unreviewed Safety Question
WAC	Waste Acceptance Criteria
WSRC	Westinghouse Savannah River Company

5.0 DERIVATION OF TECHNICAL SAFETY REQUIREMENTS

5.1 INTRODUCTION

5.1.1 OBJECTIVE

The purpose of this chapter is to provide information that addresses the requirements of Department of Energy (DOE) Order 5480.23, paragraph 8.b.(3)(p), as amplified in Attachment 1, paragraph 4.f.(3)(d)16, of the Order (Ref. 1, 2). This chapter builds upon the control functions determined to be essential in Chapters 3 and 4 and provides the information necessary for preparing the Technical Safety Requirement (TSR) document required by DOE Order 5480.22 (Ref. 3). This chapter consists of summaries and references to pertinent sections of Chapters 2, 3, and 4 in which design and administrative control features are needed to prevent or mitigate the consequences of an accident. The Safety Limits (SLs), Limiting Control Settings (LCSs), Limiting Conditions for Operation (LCOs), Surveillance Requirements (SRs), and necessary Administrative Controls determined in this chapter form the basis for the facility TSR and provide the logical link between the TSR and the Safety Analysis Report (SAR).

5.1.2 SCOPE

Expected products of this chapter include the following:

- Information with sufficient basis to derive, as appropriate, any of the following TSR parameters:
 - SLs
 - LCSs
 - LCOs
 - SRs
- Information with sufficient basis to derive TSR Administrative Controls (including specific program requirements) or to specify programs necessary to perform institutional safety functions
- Identification of all passive design features addressed in the SAR (credited in the hazards or accident analysis)
- Identification of TSRs from other facilities that affect the facility's safety basis
- Derivation of facility modes

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5.2 REQUIREMENTS

The objective of this section is to identify the DOE Orders, standards, and regulations that are required to establish the safety basis of the facility. The primary requirements applicable to the derivation of TSRs are contained in the following documents (Ref. 1 through 6):

- DOE Order 5480.23, Nuclear Safety Analysis Reports
- DOE-STD-3009-94, Preparation Guide for U.S. Department of Energy Non-Reactor Nuclear Facility Safety Analysis Reports
- DOE Order 5480.22, Technical Safety Requirements
- Westinghouse Savannah River Company (WSRC) Procedure Manual 11Q, Westinghouse Savannah River Company Facility Safety Analysis Manual (U)
- WSRC-TR-94-0466, Safety Analysis Report Preparation Guidelines
- WSRC-TM-93-9, WSRC Functional Classification Methodology Manual

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5.3 TSR COVERAGE

5.3.1 FACILITY BACKGROUND

This section presents the technical basis and rationale used in developing the TSR document for the Receiving Basin for Offsite Fuels (RBOF) and the Resin Regeneration Facility (RRF). The RBOF was constructed in the early 1960s to provide safe storage and handling of a variety of nuclear fuel types. The RRF was built at approximately the same time and is considered an integral part of the RBOF. The mission of the facility has not changed much since the facility's original construction. The mission of the facility is as follows:

- Receive, handle, and store irradiated nuclear fuel from foreign and domestic reactors, and fuel/targets from onsite reactors
- Repackage nuclear fuel elements and targets into containers or bundles necessary for extended storage and/or shipment to onsite or offsite facilities
- Maintain the provisions to receive resins used in the RBOF and resins shipped from onsite reactor facilities, to regenerate the resins, and to transfer the waste to the tank farm
- Perform each of the above functions safely with respect to the onsite worker, public, and environment through adherence to safety management programs

5.3.2 TSR METHODOLOGY

5.3.2.1 Selection of TSR Candidates

Candidate TSR controls were selected for the following items:

- Items required to prevent or mitigate the uncontrolled release of radiological or toxicological material that affects public safety
- Items required to prevent or mitigate the uncontrolled release of radiological or toxicological material that affects worker safety
- Items required to provide the required monitoring and alarm functions to detect and report an inadvertent criticality
- Items required to provide layers of protection proportional to the hazard potential to ensure that multiple features are relied upon for the prevention or mitigation of uncontrolled releases of hazardous material to the environment (i.e., defense in depth); (these equipment and administrative control features include integrated safety management programs that provide controlled and disciplined operations)

For the purposes of the derivation of TSR controls, WSRC has defined an “uncontrolled release” as any release that results from an event in which the effects could challenge or exceed the Evaluation Guidelines (EGs) as identified in Sections 5.3.2.3 (for the public) and 5.3.2.4 (for the workers). The EGs provide a measure from which management can assess the risk of an uncontrolled material release.

The following sections detail the TSR candidate selection process in the four areas previously described.

5.3.2.2 Events Affecting Public Safety (Offsite Evaluation Guidelines)

If any credible accident (an accident with a conservatively estimated frequency of greater than $1.0\text{E-}06$ events/yr) results in consequences above the offsite EGs, TSR control is required to meet the public safety criterion.

If any accident with a frequency of less than $1.0\text{E-}06$ events/yr results in unacceptable consequences (above the offsite EGs or undetermined consequences), TSR control is required for any assumptions (i.e., initial conditions or operability of a Structure, System, or Component [SSC]) required to maintain the accident frequency at less than $1.0\text{E-}06$ events/yr.

The EGs provide a measure from which the risk of an uncontrolled release of hazardous material can be assessed. Design Basis Accidents (DBAs) shall not cause a committed effective dose equivalent to the maximally exposed offsite individual exceeding the following EGs (Ref. 6):

- 0.5 rem for accidents with estimated frequencies $<1.0\text{E-}01$ events/yr but $\geq 1.0\text{E-}02$ events/yr
- 5 rem for accidents with estimated frequencies $<1.0\text{E-}02$ events/yr but $\geq 1.0\text{E-}04$ events/yr
- 25 rem for accidents with estimated frequencies $<1.0\text{E-}04$ events/yr but $\geq 1.0\text{E-}06$ events/yr

Note: The 25-rem limit is based on information provided in Title 10 Code of Federal Regulations (CFR) Part 100 and DOE Order 6430.1A; the 5-rem limit is based on information provided in 10 CFR 72; and the 0.5-rem limit is based on information provided in DOE Order 5400.5 (Ref. 7).

The nonradiological toxic EGs for selecting TSR controls for protecting the public include the following ambient concentrations (Ref. 6):

- Permissible Exposure Limit - Time-Weighted Average or $2.0\text{E-}06$ times the Incremental Cancer Risk (ICR) for accidents with estimated frequencies $<1.0\text{E-}01$ events/yr but $\geq 1.0\text{E-}02$ events/yr

- Emergency Response Planning Guideline - Level 1 (ERPG-1) or $2.0\text{E-}04$ times the ICR for accidents with estimated frequencies $<1.0\text{E-}02$ events/yr but $\geq 1.0\text{E-}04$ events/yr
- ERPG-2 or $2.0\text{E-}02$ times the ICR for accidents with estimated frequencies $<1.0\text{E-}04$ events/yr but $\geq 1.0\text{E-}06$ events/yr

5.3.2.3 Worker Safety (Onsite Evaluation Guidelines)

If any credible accident (an accident with a conservatively estimated frequency of greater than $1.0\text{E-}06$ events/yr) results in consequences above the onsite EGs, TSR control is required to meet the worker safety criterion.

If any accident with a frequency of less than $1.0\text{E-}06$ events/yr results in unacceptable consequences (above the onsite EGs or undetermined consequences), TSR control is required for any assumptions (i.e., initial conditions or operability of an SSC) required to maintain the accident frequency at less than $1.0\text{E-}06$ events/yr.

The EGs provide a measure from which the risk of an uncontrolled release of hazardous material can be assessed. DBAs shall not cause a committed effective dose equivalent to the workers at the following three receptor locations exceeding the EGs:

- The workers that are in the immediate area of the hazard (including workers that are working on or standing adjacent to the hazard)
- The workers that are within the room or building associated with the hazard (these workers are not necessarily associated with operating or maintaining the hazard)
- The workers that are outside of the last possible physical means of containing the hazard (including all workers outside of the process building containing the hazard or outside of an exclusion zone associated with a hazard that is not within a room or building, such as an open tank in the tank farm)

The EGs for TSR controls for the workers, as defined above, are as follows (Ref. 6):

- TSR controls are required when the event consequences could result in a prompt worker fatality or acute injury that would require medical treatment for immediately life-threatening or permanently disabling injuries from hazards other than those encountered in general industry and construction. (Note: this criterion, which is qualitative in nature, also applies to the worker receptors listed in this section.)
- For workers within the room or building associated with the hazard, TSR controls are required to prevent an exposure to the workers (as the result of the process component failure) when the concentration of hazardous material in an occupied area, as determined by uniform distribution of the total released material in the occupied area,

would exceed the ERPG-3 concentration limit or would exceed Hazard Category 3 thresholds for radionuclides.

- For the workers outside of the last possible means of containing the hazard, TSR controls are required, based on quantitative analysis, if the following EGs are exceeded:
 - 25 rem (prompt-radiological) or ERPG-2 (hazardous material concentration) for accidents with estimated frequencies $\geq 1.0\text{E-}04$ events/yr
 - 100 rem (prompt-radiological) or ERPG-3 (hazardous material concentration) for accidents with estimated frequencies $< 1.0\text{E-}04$ events/yr but $\geq 1.0\text{E-}06$ events/yr

5.3.2.4 Criticality

In addition to protecting the public and workers, TSR controls shall be required to prevent, monitor, or detect a criticality even if that accident would not directly impact worker safety. TSR controls are required if the criticality alarm system is required as defined in ANSI/ANS-8.3-1986 (Ref. 8). If criticality alarm equipment is not required by this standard, then no TSR controls are required for this selection item. The alarm system must cover occupied areas in which the expected dose exceeds 12 rad in free air. An alarm system is not required for a storage basin if sufficient shielding to protect personnel (dose from initial burst < 12 rad) exists, but a fission product detection system should be provided in adjacent occupied area. Credible criticalities in the RBOF basin will not result in a dose to personnel inside or outside the building exceeding 12 rads. Therefore, exception can be taken to the need for a NIM system in the RBOF. The NIM system functions as a fission product detection system.

5.3.2.5 Defense in Depth

TSR controls are also selected to provide layers of protection proportional to the hazard potential. These additional controls ensure that multiple features are relied upon for the prevention or mitigation of the uncontrolled release of hazardous material and provide a defense in depth. Levels of control are selected to provide defense-in-depth protection in the TSRs. A level of control is defined as “a structure, system, component, Administrative Control, natural phenomena, or inherent feature which can be readily expected to act to prevent or mitigate the release of hazardous material to an unwanted location” (Ref. 6). Due to the depth of design at Savannah River Site (SRS) facilities, there are usually more levels of control available than the minimum requirements necessary to meet the EGs. Not all of the controls identified in the facility description of the SAR are required to be included in the TSRs. The number of controls selected to be included in the TSR are based on the significance of the hazard and are determined based on the following guidance.

TSR controls shall be placed on defense-in-depth SSCs or Administrative Controls according to the following minimum level-of-control criteria (Ref. 6):

- If any credible event has consequences that are greater than ERPG-3 or 100 rem or that exceed the offsite guidelines, two levels of control are required.
- If any credible event has consequences that are greater than ERPG-2 or 25 rem (for the workers only), one level of control is required.

For the receptors (for the workers and the public) outside the last possible means of containing the hazard, additional TSR controls shall be placed on defense-in-depth SSCs or Administrative Controls according to the following minimum level-of-control criteria (Ref. 6):

- An additional level of control is required if there is sufficient energy in the process material to result in an unacceptable event (e.g., explosion).
- An additional level of control is required if the geometry of the hazardous material must be maintained (e.g., geometry of radiological material maintained to prevent critical mass; two reactive chemicals separated to prevent explosion).
- When additional levels of control are required, they should first be selected to avert damage to the plant and to the existing barriers themselves; secondly, they should be selected to add additional barriers.

5.3.3 FACILITY EVALUATION

The TSR selections for the RBOF and RRF were based upon events analyzed in the hazards and accident analysis (Chapter 3). The events or accidents analyzed met the following criteria:

- Hazards analysis events that were screened positive in the defense-in-depth evaluation of the Functional Classification Report (Ref. 9) (affecting worker and/or public safety)
- Accident analysis events that resulted in the release of radiological or toxicological material exceeding the offsite risk guidelines (affecting public safety)
- Accident analysis events that resulted in inadvertent criticality (Ref. 9, 10)
- Accident analysis events that would otherwise meet criterion b or c if not for an initial assumption (i.e., initial conditions or operability of an SSC) that affects either the prevention or mitigation of the event (Ref. 9)

5.3.3.1 Accidents/Events Evaluated in the DBA Analysis

The accidents analyzed in Section 3.4 (quantitative) are derived from the hazards analysis summarized in Section 3.3. Potential accident events identified in the hazards analysis were binned into risk categories based on likelihood of occurrence and consequence. A bounding

analysis is performed for all accident types of concern, unless the accident is bounded by a unique accident analysis. Accidents analyzed include the following groups:

Unique Accidents (as defined by DOE-STD-3009-94) (Ref. 2)

1. Chemical spills
 - a. Loss of 50% sodium hydroxide containment (Event 316)
 - b. Loss of 50% nitric acid containment (Event 318)
 - c. Loss of 30% sodium nitrite containment (Event 314a)

2. Portable deionizer resin fire (Event 310)
3. Uncontrolled chemical reaction (Event 51)

Representative Bounding Accidents (as defined by DOE-STD-3009-94)

1. Earthquake (Event 43)
2. High winds/tornado (Event 63, 63a, and 64)
3. Temperature extremes (Event 348)
4. Surface vehicle impact (Events 345c and 345d are bounded by Events 314a, 316, and 318)
5. Aircraft impact (Event 62)
6. Structural failure (Event 63, 63a, and 64)
7. Missile impact (Events 314a, 316, and 318)
8. Criticality (Events 34, 35, 37b, 38, 39, 40, 40a, 41, 42, and 44)
9. Explosion (Event 400a)
10. Fire (Event 310)
11. Uncontrolled chemical reaction (Event 51)
12. Active loss of confinement (Events 314a, 316, and 318)
13. Passive loss of confinement (Events 314a, 316, and 318)

CHEMICAL SPILLS

Events 314a, 316, and 318 involve chemical spills that have the potential to release chemicals in excess of the ERPG-2 values, which are of moderate consequences to the workers. The Hazardous Material Inventory Control Program limits the type (e.g., sodium nitrite, sodium hydroxide, and nitric acid) and amount of chemical inventory available for release. The designs of the chemical pad No. 2 berm, and drain system to waste tank EP38 ensure that sodium nitrite does not mix with nitric acid. Trained workers follow operating procedures for the chemical pad

No. 2 pump, which prevents pumping the contents of chemical pad No. 2 to waste tank EP38 without first sampling for nitric acid. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). These TSR controls are described in Table 5.3.1-1. Also, workers are trained to attempt to limit the spill size and clean up the spill in a timely fashion; however, this spill response is considered defense in depth.

SURFACE VEHICLE IMPACT

Events 345c and 345d involve surface vehicles causing damage to the chemical tanks resulting in spills similar to Events 314a, 316, and 318 with similar consequences. The Hazardous Material Inventory Control Program limits the type (e.g., sodium nitrite, sodium hydroxide, and nitric acid) and amount of chemical inventory available for release. Also, the facility design is such that the chemical tanks are remote to the normal traffic pattern. The chemical pad No. 2 location will ensure that the tank is not exposed to normal vehicles traffic. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). These TSR controls are described in Table 5.3.1-2.

PROCESS-INDUCED CRITICALITY

Events 34, 35, 36, 37a, 37b, 38, 39, 40, 40a, 41, 42a, and 44 involve postulated criticalities that were determined to be of moderate consequence to the worker. Events 24, 40b, 42, and 404 were determined to have a frequency of occurrence in the beyond extremely unlikely range. The following equipment and Administrative Controls are used to ensure that the frequency of a process-induced criticality is as analyzed in Chapter 3 or as assumed in the hazards analysis:

- Storage racks and associated hardware to provide spacing.
- Bundles, buckets, bucket inserts, and blocking devices protect the spacing assumptions of the criticality analyses.
- Workers are trained in the safe operation of the traveling bridge and safe handling of fuel elements, assemblies, and bundles.
- The Nuclear Criticality Safety Program specifies controls to prevent fuel canning infractions, rack or fuel snagging, bundling infractions, fuel drops and spills, and unauthorized disassembly operations. Also, the Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. The NSDSs define the allowable storage configurations and storage hardware, as well as the allowable neutronic interaction with other nearby fuels.
- Criticality control requirements control fuel handling and storage in accordance with the applicable NSDS. The following also serve to reduce the frequency of this event:

fuel storage in isolation, neutron reflectors, support grating, and cask basket unloading controls.

- The Fuel Receipt and Shipping Program ensures the following:
 - Casks used to transfer the fuel have been analyzed and loaded in a manner that is critically safe.
 - The casks have documentation (offsite: Safety Analysis Report for Packaging and Certificate of Compliance, onsite: WSRC Procedure Manual 19Q [Ref. 11]).
 - Fuel and its containers have sufficient negative buoyancy.
 - Characteristics of the fuel received meet the assumptions in the criticality analyses.
 - Number of casks received, fuel elements received, shipments from the RBOF, number of operations, etc., are tracked or are sufficiently bounding to meet the assumptions of the frequency analyses of the criticality events.
 - Fuel is received within the maximum heat load to ensure that the fuel will not melt if stored in a normal storage configuration.
 - Fuel integrity must be confirmed prior to shipping fuel to the RBOF.
 - Casks have been analyzed for deflagration concerns.
 - Fuel does not contain any materials that react violently with water (e.g., sodium or potassium).
 - Fuel meets storage rack weight requirements of the structural analysis.
 - Casks containing antifreeze are not purged to waste tank EP38.
- The Basin Water Chemistry Control Program ensures that the chemistry of the basin is favorable for long-term storage of fuel and that only minimal corrosion of the fuel occurs. In addition, the program ensures that the alpha activity of the basin water remains within the analyzed amount such that fissile materials will not accumulate to a critical mass in the basin water filtration and deionizer system.
- The Deionizer Resin Replacement/Regeneration Program ensures that resins are changed before a critical mass of fissile material is captured by the resins.
- The Research Reactor Fuel Counter (RRFC) safety skirt is installed to prevent fuel being inadvertently placed below the RRFC.
- The 100-ton crane south travel limit switch system and mechanical stop prevent the crane from carrying and dropping a load into a fuel storage basin or from traveling beyond the end of the crane rails. Also, the crane operating procedures direct workers to park the crane in a safe position and to use adequate caution when making lifts.
- The Hoisting and Rigging Program ensures that the crane and its associated equipment is maintained which minimizes its likelihood of failure.

- SRS 70-ton cask is designed to provide containment of materials in the cask.
- SRS 70-ton cask shipping requirements ensure that cadmium plates and cask slot blocking devices are installed in accordance with the applicable NSDS. Also, the requirements ensure that the cadmium plates have passed a neutron transmission test.

If a postulated process-induced criticality were to occur, the following equipment and Administrative Controls would mitigate the consequences to the workers and public:

- The Fuel Receipt and Shipping Program ensures that the fuel characteristics meet the assumptions used in the analyses of a process-induced criticality.
- The cask/fuel receiving requirements ensure that the fuel identification matches the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses.
- Nuclear Incident Monitors (NIMs) alert workers that a criticality with a fission product release is in progress. (The NIM trouble portion of the remote 244-H general alarm and the remote NIM alarm are included for this function.)
- Basin water provides shielding to workers in the area of the criticality and a decontamination factor for fission products released as a result of a criticality.
- The basins and canals provide containment of the basin water and structural support for the storage racks and associated hardware.
- Basin walls provide shielding to workers in the vicinity of the basin.
- The control room concrete wall provides shielding to workers from direct radiation.
- Workers are trained to properly respond in the event of a criticality.
- The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).
- The Radiological Protection Program provides for monitoring of evacuating workers and performance of dose assessments.

These TSR controls are described in Table 5.3.1-3.

SEVERE WEATHER - CHEMICAL SPILLS

Events 347 and 348 involve chemical spills caused by either high wind-, earthquake-, or snow- and ice-induced rain shelter failure causing damage to the chemical tanks resulting in spills. The Hazardous Material Inventory Control Program limits the type (e.g., sodium nitrite, sodium hydroxide, and nitric acid) and amount of chemical inventory available for release. The designs of the chemical tank (e.g., sodium nitrite, nitric acid), chemical pad No. 2 berm, and drain system to waste tank EP38 ensure that sodium nitrite does not mix with nitric acid (see Event 401a).

Trained workers follow operating procedures for the chemical pad No. 2 pump, which prevents pumping the contents of chemical pad No. 2 to waste tank EP38 without first sampling for nitric acid. Also, trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). These TSR controls are described in Table 5.3.1-4.

SEVERE WEATHER - HIGH WIND-INDUCED CRITICALITY

Events 63, 63a and 64 involve high wind-induced criticalities caused by RBOF building components falling into the basin and dislodging fuel from the racks into a critical configuration. These events were determined to be of moderate consequences to the workers. The following equipment and Administrative Controls are used to reduce the frequency of a high wind-induced criticality:

- Storage racks and associated hardware to provide spacing.
- Bundles, buckets, bucket inserts, and blocking devices protect the spacing assumptions of the criticality analyses.
- The 244-H building design withstands winds according to the fragility curve used in the frequency analysis for this event.
- Criticality control requirements control fuel handling and storage in accordance with the applicable NSDS. The following also serve to reduce the frequency of this event: fuel storage in isolation, neutron reflectors, support grating, and cask basket unloading with fuel units greater than the applicable safe number.
- The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. The NSDSs define the allowable storage configurations and storage hardware, as well as the allowable neutronic interaction with other nearby fuels.
- The Fuel Receipt and Shipping Program ensures the following:
 - The casks have documentation (offsite: Safety Analysis Report for Packaging and Certificate of Compliance, onsite: WSRC Procedure Manual 19Q [Ref. 11]).
 - Fuel meets storage rack weight requirements of the structural analysis.

If a postulated high wind-induced criticality were to occur, the following equipment and Administrative Controls would mitigate the consequences to the workers and the public:

- The Fuel Receipt and Shipping Program ensures that the fuel characteristics meet the assumptions used in the analyses of a high wind-induced criticality.
- The cask/fuel receiving requirements ensure that the fuel identification matches the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses.

- Basin water provides shielding to workers in the area of the criticality and a decontamination factor for fission products released as a result of a criticality.
- The basins and canals provide containment of the basin water and structural support for the storage racks and associated hardware.
- Basin walls provide shielding to workers in the vicinity of the basin.
- The control room concrete wall provides shielding to workers from a criticality that occurs outside of the basin.
- Workers are trained to properly respond in the event of a criticality.
- The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).
- The Radiological Protection Program provides for monitoring of evacuating workers and performance of dose assessments.

These TSR controls are described in Table 5.3.1-4.

SEVERE WEATHER - FUEL BREACH (EARTHQUAKE AND HIGH WIND-INDUCED)

Event 18 involves a fuel breach, which is postulated to occur as a result of an NPH event, causing the facility to fail with debris falling into the storage basin and rupturing fuel bundles. A fuel breach was postulated to result in low consequences to the workers. The following equipment and Administrative Controls are used to reduce the frequency of an NPH-induced fuel breach:

- The 244-H building design withstands an NPH event according to the fragility curve used in the frequency analysis for this event.
- Storage racks and associated hardware to provide spacing.
- Bundles, buckets, bucket inserts, and blocking devices protect the spacing assumptions of the criticality analyses.
- The Fuel Receipt and Shipping Program ensures that the fuel meets storage rack weight requirements of the structural analysis.
- Criticality control requirements control fuel handling and storage in accordance with the applicable NSDS. The following also serve to reduce the frequency of this event: fuel storage in isolation, neutron reflectors, support grating, and cask basket unloading with fuel units greater than the applicable safe number.
- The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. The NSDSs define the allowable storage configurations and storage hardware, as well as the allowable neutronic interaction with other nearby fuels.

The following equipment and Administrative Controls are used to mitigate an NPH-induced fuel breach:

- The Fuel Receipt and Shipping Program ensures that the fuel characteristics meet the assumptions used in the consequence analysis of a NPH-induced fuel breach.
- The cask/fuel receiving requirements ensure that the fuel identification matches the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses.
- Basin water provides protection to the fuel from missiles that strike the water. These missiles are slowed by the water, potentially preventing damage by the missiles.
- The basins and canals provide containment of the basin water and structural support for the storage racks and associated hardware.
- Workers are trained to properly respond in the event of a fuel breach.
- The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).
- The Radiological Protection Program provides for monitoring of evacuating workers and performance of dose assessments.

These TSR controls are described in Table 5.3.1-4.

EARTHQUAKE - NPH-INDUCED CRITICALITY

Event 63 involves a criticality caused by natural phenomena. Event 63 postulates an NPH event causing the RBOF building structure to fall into the basin deforming the racks and resulting in the fuel being reconfigured into a critical mass. This event was determined to be of high or moderate consequences to the workers.

The following equipment and Administrative Controls are used to reduce the frequency of an earthquake-induced criticality:

- Storage racks and associated hardware to provide spacing.
- Bundles, buckets, bucket inserts, and blocking devices protect the spacing assumptions of the criticality analyses.
- The 244-H building design withstands earthquakes according to the fragility curve used in the frequency analysis for this event.
- The basins and canals are assumed to withstand earthquakes allowing the storage racks and hardware to remain intact.
- Criticality control requirements control fuel handling and storage in accordance with the applicable NSDS. The following also serve to reduce the frequency of this event:

fuel storage in isolation, neutron reflectors, support grating, and cask basket unloading with fuel units greater than the applicable safe number.

- The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. The NSDSs define the allowable storage configurations and storage hardware, as well as the allowable neutronic interaction with other nearby fuels.
- The Fuel Receipt and Shipping Program ensures the following:
 - The casks have documentation (offsite: Safety Analysis Report for Packaging and Certificate of Compliance, onsite: WSRC Procedure Manual 19Q [Ref. 11]).
 - Fuel meets storage rack weight requirements of the structural analysis.
- The cask/fuel receiving requirements ensure that the fuel identification matches the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses.
- The 100-ton crane south travel limit switch system and mechanical stop prevent the crane from carrying and dropping a load into a fuel storage basin or from traveling beyond the end of the crane rails. The Hoisting and Rigging Program ensures that the crane and its associated equipment is maintained which minimizes its likelihood of failure. Also, the crane operating procedures direct workers to park the crane in a safe position and to use adequate caution when making lifts.

By assuming that this equipment and Administrative Controls reduce the frequency of an earthquake, the earthquake-induced criticality was determined to be beyond extremely unlikely. Therefore, no mitigators are required for this event.

These TSR controls are described in Table 5.3.1-5.

EARTHQUAKE - BREACH OF POOL BASIN

Events 43 and 63a involves a postulated earthquake-induced breach of the RBOF basin and subsequent draindown. These events were determined to be of negligible consequences to the workers. This event is prevented by the earthquake design of the basins and canals, which contain the basin water and only allow minor leakage. The basin water that does leak from the basin results in low consequences based on the inventory controls of the Basin Water Chemistry Control Program. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). These TSR controls are described in Table 5.3.1-5.

EARTHQUAKE - BREACH OF WASTE TANK EP38

Event 401 involves a postulated earthquake-induced breach of waste tank EP38 and subsequent airborne radionuclide release. The inventory in waste tank EP38 is controlled to the assumed value of the consequence analysis by the Deionizer Resin Replacement/Regeneration Program. The deionizer resin is assumed to have the worst-case liquid radionuclide inventory in the facility. The Radiological Protection Program provides for monitoring of evacuating workers and performance of dose assessments. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). Also, the response procedures requires that the waste is cleaned up within 7 days as assumed in the consequence analysis of this event. Workers are trained to properly respond following an earthquake. These TSR controls are described in Table 5.3.1-5.

EARTHQUAKE - LOSS OF CHEMICAL DUMPSTER CONTAINMENT

Event 347 involves an earthquake event causing a chemical dumpster breach resulting in spills. The Hazardous Material Inventory Control Program limits the type (e.g., sodium nitrite, sodium hydroxide, and nitric acid) and amount of chemical inventory available for release. The designs of the chemical tanks (i.e., sodium nitrite, nitric acid), chemical pad No. 2 berm, and drain system to waste tank EP38 ensure that sodium nitrite does not mix with nitric acid (see Event 401a). Trained workers follow operating procedures for the chemical pad No. 2 pump, which prevents pumping the contents of chemical pad No. 2 to waste tank EP38 without first sampling for nitric acid. The chemical tanks were determined to survive an earthquake event. Therefore, there is no postulated spill. However, trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). These TSR controls are described in Table 5.3.1-5.

FIRE - RESIN FIRE

Event 310 postulates a resin fire involving the portable deionizers in the RRF that has the potential to release radionuclides that are of moderate consequences to the workers. The Reactor Deionizer Resin Replacement/Regeneration Program limits the alpha content of the resin to prevent self-ignition and the radionuclide content of the resin to within the content assumed in the consequence analysis. The Radiological Protection Program provides for monitoring of evacuating workers and performance of dose assessments. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). These TSR controls are described in Table 5.3.1-6.

EXPLOSION - NITRIC ACID AND RESIN

Event 400a involves mixing undiluted nitric acid and resin, causing an explosion that has the potential to release radionuclides that are of low consequences to the workers. Both the RBOF and the Reactor Deionizer Resin Replacement/Regeneration Program limit the radionuclide content of the resin to within the content assumed in the consequence analysis. The Radiological Protection Program provides for monitoring of evacuating workers and performance of dose assessments. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). These TSR controls are described in Table 5.3.1-7.

UNCONTROLLED CHEMICAL REACTION - ANY MATERIAL THAT REACTS VIOLENTLY WITH WATER (E.G., SODIUM AND POTASSIUM)

Event 51 involves a violent chemical reaction between offsite fuel that contains any material that reacts violently with water (e. g., sodium and potassium). Fuel is postulated to be delivered to the RBOF with sodium not stripped properly. This violent reaction has the potential to release radionuclides that are of moderate consequences to the workers. This event was identified by the hazards analysis as requiring unique accident analyses. The Fuel Receipt and Shipping Program prevents exposed material (e.g., sodium and potassium) which would react violently with water from being delivered to the RBOF. The cask/fuel receiving requirements ensure that the unloaded fuel is the fuel described in the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses or is isolated. If the violent reaction occurs, radiation monitoring alert workers to the release of radionuclides. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). These TSR controls are described in Table 5.3.1-8.

AIRCRAFT IMPACT

Aircraft impact with the facility was analyzed in Chapter 3. The frequency of this accident was shown to be beyond extremely unlikely. No TSR controls are required to protect this frequency. This accident is addressed in Table 5.3.1-9.

5.3.3.2 Hazard Events Not Included in the Design Basis Accident Analysis

Several events were identified during the screening process performed for defense in depth and functional classification that could affect worker protection and are not included in the bounding DBA events. The hazards analysis evaluated and screened the hazard events to determine which events posed a concern to the public and/or worker. The events that form a concern for worker safety not included in the bounding DBA events are evaluated and discussed below. The following sections discuss these events and specifically show the recommended TSR controls for

the evaluated events. Tables 5.3.2-1 through 5.3.2-10 summarize these recommended TSR controls.

INJURY TO PERSONNEL - GROUP 1 FIRE/EXPLOSION

Controls for the following events have been selected and documented in Table 5.3.2-1.

Events 313 and 314

These worker injury events (Events 313 and 314) involve explosions/fires that have the potential to release chemicals in excess of the ERPG-2 values and are of moderate consequences. The Hazardous Material Inventory Control Program limits the chemical inventory. The Fire Protection Program ensures that appropriate fire protection measures are maintained (limiting combustibles, fire preplans, etc.). Also, trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 400

Event 400 involves the potential for a fire/explosion in waste tank EP38 as a result of coolant (containing flammable antifreeze) of a received cask being discharged to the waste tank. Based on calculations performed in Section 3.3.2.3, an explosion in waste tank EP38 could result in a worker fatality. The Fuel Receipt and Shipping Program prohibits the transfer of cask contents containing flammable antifreeze to the waste tank. The Hazardous Material Inventory Control Program also prohibits the storage of flammable antifreeze (e.g., ethylene glycol, methanol) at any location where it could drain to waste tank EP38. The Fire Protection Program ensures that appropriate fire protection measures are maintained (limiting combustibles, fire preplans, etc.) and that the safe condition of the facility is maintained following a fire or explosion. If a transfer were to occur, workers are assumed to follow their training by reporting the error and evacuating. The Radiological Protection Program ensures that the monitoring is available, provides for monitoring of evacuating workers and performance of dose assessments. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Events 2a and 3

Events 2a and 3 involve explosion/fires that could potentially affect casks. These events are predicted to have low consequences, assuming the containment design for the cask is maintained. Event 3 involves a minor explosion from radiolytic decomposition resulting in hydrogen gas production. The Fuel Receipt and Shipping Program controls the design of casks and flammable gas generation concerns for known failed fuel assemblies. Offsite casks are fire rated in accordance with 10 CFR 71 (Ref. 12). The Fire Protection Program ensures that appropriate fire

protection measures are maintained (limiting time tractor is in carport, limiting combustibles, fire preplans, etc.) and that the safe condition of the facility is maintained following a fire or explosion. Also, the trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Events 310 and 400b

Events 310 and 400b involve a fire in the resin. Due to the potential radioactivity of the deionizer cation resin, a fire has the potential to expose the workers to greater than 25 rem. Event 310 was analyzed as a DBA in Section 3.4.2.6 and is discussed in Section 5.3.3-1. The analysis of event 400b assumes that the Basin Water Chemistry Control Program and Deionizer Resin Replacement/Regeneration Program limits the alpha content of the water and resin so that likelihood of self-ignition of the resin is minimized. The Radiological Protection Program ensures that the monitoring is available and provides for monitoring of evacuating workers and performance of dose assessments. Also, the trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

INJURY TO PERSONNEL - GROUP 2 CONTAINMENT/SURFACE CONTAMINATION

Controls for the following events have been selected and documented in Table 5.3.2-2.

Events 4 and 23

Events 4 and 23 involve receiving casks damaged or deficient that could result in the spread of contamination. These events result in low consequences to the workers, assuming the containment design for casks is maintained and a reasonable worst-case contamination level of less than 1 Ci is maintained. Also, the Radiological Protection Program ensures that casks are surveyed when received. The trained workers will follow the instructions of the Radiological Control Operations (RCO) personnel to avoid contacting and spreading the contamination. Assuming the Fuel Receipt and Shipping Program protects the source term and the containment design of the casks assumed in the hazards analysis, these events result in low consequences.

Event 9

Event 9 involves the potential for a basin overflow. Assuming a reasonable worst-case water activity level of 0.1 $\mu\text{Ci/ml}$ and an overflow of 75,600 liters, the total curies involved in the overflow result in low consequences. The Basin Water Chemistry Control Program protects this assumption. The Radiological Protection Program would require that the spill be monitored to determine the activity of the spilled water. The trained workers follow the applicable event

response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

INJURY TO PERSONNEL - GROUP 2 CONTAINMENT/ SUBSURFACE RELEASES

Controls for the following events have been selected and documented in Table 5.3.2-2.

Events 8, 8a, and 8b

Events 8, 8a, and 8b involve the releases of basin water. Event 8 involves a leak from the basin to the subsoil from a dropped cask or earthquake. Assuming the basin water activity is maintained, the results of this event are negligible. The Basin Water Chemistry Control Program protects this assumption. Also, the basin design prevents a significant leak from occurring in the basin and provides structural support for storage racks and associated hardware. Event 8b involves a leak from the filter - deionizer system. In addition to the Basin Water Chemistry Control Program. The configuration of the filter and deionizer pump (limited pump inlet depth) prevents a significant leak from occurring in the basin. Also, Event 8a considers the impact that a basin draindown would have on fuels presently stored in the basin. Based on analysis, it has been determined to be beyond extremely unlikely for a fuel element to melt, assuming the maximum heat load is controlled and the normal storage configuration is maintained.

The Fuel Receipt and Shipping Program ensures that the maximum heat load of fuel is sufficiently low to prevent fuel melting in air if the fuel is stored in a normal storage configuration. The cask/fuel receiving requirements ensure that the unloaded fuel is the fuel described in the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses or is isolated. The hoisting and rigging program ensures the crane and related equipment is available for use and minimizes the likelihood of failure of the components and load dropping. Fuel is assumed to be stored properly in accordance with the criticality control requirements and the applicable NSDS. The designs of the storage rack, hardware configuration, buckets, bucket inserts, blocking devices, and bundles maintain the stored fuel configuration, which prevents fuel interaction causing fuel damage. The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. The NSDSs define the allowable storage configurations and storage hardware, as well as the allowable neutronic interaction with other nearby fuels.

The Radiological Protection Program will provide coverage when workers are around the basin during the draindown event and would indicate to the workers when conditions warranted them to leave. In addition, it is assumed that if these events cause a NIM to alarm (reduced water level causing the basin area to become a high radiation area), the trained workers would immediately evacuate the area, thereby mitigating the consequence. To prevent workers from entering into the basin area with the basin in a draindown condition during shutdown, the NIM remote alarm and trouble alarms are assumed to be operating. The trained workers follow the applicable event

response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 321

Event 321 involves the potential breach of a portable deionizer or its associated piping, hoses, and valves, and basin water and contaminated resin leaking to the subsurface. Assuming a worst-case activity for the basin water and resin, it was concluded that this event results in low consequences. The Reactor Basin Water Chemistry Control Program and Reactor Deionizer Resin Replacement/Regeneration Program control this assumption. The Radiological Protection Program would require that the spill be monitored to determine the activity of the spilled resin. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 406

Event 406 involves the potential breach of a waste transfer line with a known leak in the secondary transfer line containment. Assuming a worst-case activity for the waste being transferred, it was concluded that this event results in a negligible consequence. The Radiological Protection Program will provide coverage when workers are around the waste transfer line and would indicate to the workers when conditions warranted them to leave. Trained workers are assumed to follow the Radiological Protection Program when transferring the waste. Also, the design of the transfer line (i.e., underground with a secondary containment) protects the workers from exposure.

INJURY TO PERSONNEL - GROUP 2 CONTAINMENT/AIRBORNE RELEASES

The quantity of radionuclides at risk for events involving spent fuel is based on a worst-case fuel assembly (i.e., Reference Fuel Assembly [RFA]) with characteristics described in Section 3.3.1.1. The RFA is bounding and is therefore used as the reference fuel when evaluating consequences involving irradiated fuel assemblies. The Fuel Receipt and Shipping Program and cask/fuel receiving requirements control this assumption.

Controls for the following events have been selected and documented in Table 5.3.2-2.

Events 5, 6, 12, and 13

Events 5, 6, 12, and 13 involve cask handling operations. Assuming the containment design of the cask and the radiological source term are controlled, the consequences of these events are low. The Fuel Receipt and Shipping Program controls the containment design of the cask and the source term in a cask (no known leaking fuel is placed in cask without canning or isolation). The Radiological Protection Program provides radiation monitoring and ensures that trained

workers evacuate upon a high radiation condition. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 5a

Event 5a involves cask handling of onsite casks. Assuming the containment design of the cask and the radiological source term are controlled, as well as workers' response to radiation monitoring, the results of this event are low. The Fuel Receipt and Shipping Program controls the containment design of the cask and the source term in a cask (no known leaking fuel is placed in cask without canning or isolation). The Radiological Protection Program provides radiation monitoring and ensures that trained workers evacuate upon a high radiation condition. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Events 7, 7a, 10, 10a, 11, and 11a

Events 7, 7a, 10, 10a, 11, and 11a involve fuel or target handling operations and storage where the potential exists for fuel cladding to be damaged or cut or ruptured, resulting in the release of radioactive gases to the basin. Assuming the fuel characteristics are controlled (within the RFA or assumed target assumptions), as well as workers' response to radiation monitors, the result of these events has been determined to be low. The Fuel Receipt and Shipping Program controls the source term. The cask/fuel receiving requirements ensure that the unloaded fuel is the fuel described in the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses. The Nuclear Criticality Safety Program controls cropping operations. The Radiological Protection Program provides radiation monitoring and ensures that trained workers evacuate upon a high radiation condition. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 8c

Event 8c involves the releases of basin water and direct shine from the basin caused by a dropped crane load external to the RBOF. Assuming the basin water activity is maintained and fuel does not melt, the results of this event are low. The Basin Water Chemistry Control Program protects the water activity assumption. Also, the basin design prevents a significant leak from occurring in the basin. The Hoisting and Rigging Program, along with crane lift procedure requirements, help to prevent dropping the shipping cask. Analysis shows fuel melting to be beyond extremely unlikely, assuming the maximum heat load is controlled and fuel is stored in a normal storage configuration. The Fuel Receipt and Shipping Program ensures that the maximum heat load of fuel is sufficiently low to prevent fuel melting when the fuel is stored in its normal storage configuration. The cask/fuel receiving requirements ensure that the unloaded fuel is the fuel

described in the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses or is isolated.

The Radiological Protection Program provides radiation monitoring and ensures that trained workers evacuate upon a high radiation condition. In addition, it is assumed that if this event causes a NIM to alarm, the trained workers would immediately evacuate the area, thereby mitigating the consequence. To prevent workers from entering into the basin area during shutdown, the NIM remote alarm and trouble alarms are assumed to be operating. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 324

Event 324 involves the potential breach of portable deionizers or associated piping, hoses, and valves, resulting in an airborne release to the immediate area. Assuming a worst-case activity for the portable deionizers, it was concluded that this event results in negligible consequences. The Reactor Basin Water Chemistry Control Program and Reactor Deionizer Resin Replacement/Regeneration Program control the inventory in the portable deionizers. In addition, it is assumed that if this event causes a high radiation condition, the workers would evacuate the area, thereby mitigating the consequence. The Radiological Protection Program ensures that radiation monitoring is available, and worker training ensures that the workers evacuate upon a high radiation condition. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 401a

Event 401a involves the generation of nitrous dioxide. This event postulates that nitric acid and sodium nitrite will mix in waste tank EP38. The event could be initiated by an earthquake, high winds, or tornado. It is concluded that this hazard could impact the workers. The sodium nitrite tank, sodium nitrite hold tank, and nitric acid tank are assumed to survive an earthquake event and to remain within the bermed area during a high wind or tornado event. The Hazardous Material Inventory Control Program sets the limit for the sodium nitrite inventory such that it does not overflow the berm. The designs of the chemical pad No. 2 berm and drain system provide confinement of a sodium nitrite spill. Trained workers follow operating procedures for the chemical pad No. 2 pump, which prevents pumping the contents of chemical pad No. 2 to waste tank EP38 without first sampling for nitric acid. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). However, the prevention features listed make the frequency of this event in the beyond extremely unlikely range and no mitigation features are required.

Event 402

Event 402 postulates a breach of the low-level waste trailer or other containers used to contain the liquid from the cask wash pit. Assuming a worst-case activity for the waste in the trailer (or other container), it was concluded that this event results in a low consequence. The Waste Management Program, which invokes the WAC of the receiving facility, protects the worst-case activity. Trained workers are assumed to follow the Waste Management Program when transferring the waste. The Radiological Protection Program provides radiation monitoring and ensures that trained workers evacuate upon a high radiation condition. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 407

Event 407 involves the potential transfer error of sending waste above the WAC to the general purpose evaporator in Building 211-H. Assuming a worst-case activity for the waste being transferred, it was concluded that this event results in a negligible consequence. The Waste Management Program, which invokes the WAC of the receiving facility, protects the worst-case activity. Trained workers are assumed to follow the Waste Management Program when transferring the waste.

INJURY TO PERSONNEL - GROUP 3 LOSS OF CONFINEMENT

Controls for the following events have been selected and documented in Table 5.3.2-3.

Events 16 and 17

Events 16 and 17 involve potential releases through the RBOF carport door. Assuming assemblies are bound by the RFA fuel characteristics, the resulting consequences for these events would be low. The Fuel Receipt and Shipping Program ensures that the fuel shipped to the RBOF meets the RFA fuel characteristics. The cask/fuel receiving requirements ensure that the unloaded fuel is the fuel described in the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses. During movement of a trailer into the RBOF carport, the Site Safety Program requires a spotter to alert the driver if the trailer is in danger of causing damage. In addition, the Radiological Protection Program provides radiation monitoring and ensures that trained workers evacuate upon a high radiation condition. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Events 18 and 19

Events 18 and 19 involve unfiltered ground releases caused by natural phenomena that damage the RBOF structure, resulting in fuel damage. Assuming assemblies are bound by the RFA fuel characteristics, the resulting consequences for these events would be low. The Fuel Receipt and Shipping Program ensures that the fuel shipped to the RBOF meets the RFA fuel characteristics. The cask/fuel receiving requirements ensure that the unloaded fuel is the fuel described in the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses. Fuel is assumed to be stored properly in accordance with the criticality control requirements and the applicable NSDS. The designs of the storage rack, hardware configuration, buckets, bucket inserts, blocking devices, and bundles maintain the stored fuel configuration following an earthquake event, which prevents fuel interaction causing fuel damage. Basin water level, and the basins and canal system ensure that basin water provides a protection function for the fuel from missiles and falling debris during NPH events. Also, the basins and canal system provides structural support for the storage racks and associated hardware. In addition, the trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). The Radiological Protection Program provides radiation monitoring and ensures that trained workers evacuate upon a high radiation condition.

Events 20 and 21

Events 20 and 21 involve damage, failure, or pluggage of the RBOF filters resulting in minor releases of radioactive particles. These events assume the particulate air filter fails with a reasonable worst-case release of 100 Ci of Cs-137. Comparing this source term to the minimum source term necessary to meet consequence level threshold values, it was concluded that these events result in negligible consequences to the workers and the public. Controls for the particulate filter source term are ensured through periodic ventilation inspection, which is performed as part of the Radiological Protection Program. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 329c

Event 329c involves tornado missiles causing damage to equipment in the regeneration cell releasing resins. Assuming the radionuclides in the resin are controlled, this event results in low consequences. The Deionizer Resin Replacement/Regeneration Program limits the radionuclide content of the resin to within the analysis assumptions. In addition, the trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). Worker training ensures that the workers evacuate following an NPH event. The Radiological Protection Program provides radiation monitoring and ensures that trained workers evacuate upon a high radiation condition.

When conditions have stabilized, the response procedures ensures that the spill is cleaned up in a safe and timely manner.

INJURY TO PERSONNEL - GROUP 4 INDUSTRIAL-RELATED EVENTS

The RBOF and RRF contain hazardous energy sources and hazardous material sources capable of causing injury to the workers during normal routine operations. These events are grouped into three types of events that can cause injury to workers: exposure to radiation, exposure to hazardous chemicals, and industrial events.

Controls for the following events have been selected and documented in Table 5.3.2-4.

Events 14 and 22

Events 14 and 22 involve the potential for radiological impact to the workers. These two events involve the movement of fuel into the air, either from a cask or raising the fuel above the basin water level. Assuming the radiological source term is controlled, as well as workers' response to radiation monitoring, the result of these events has been determined to be low. The Fuel Receipt and Shipping Program controls the source term. The cask/fuel receiving requirements ensure that the unloaded fuel is the fuel described in the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses. The NIMs and Radiological Protection Program provides radiation monitoring and ensures that trained workers evacuate upon a high radiation condition. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 25

Event 25 involves the potential of receiving fuel not described on the shipping manifest. Assuming all the fuel characteristics are in compliance with the Fuel Receipt and Shipping Program, the impact to the workers is low. This assumption is reasonable because of the bounding assumptions of the RFA and the fact that any fuel DOE is scheduled ship to the RBOF will be within the RFA. The Fuel Receipt and Shipping Program ensures that fuel shipped to the RBOF is described on the shipping manifest. The cask/fuel receiving requirements ensure that the unloaded fuel is the fuel described in the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses or is isolated. The Radiological Protection Program provides radiation monitoring and ensures that trained workers evacuate upon a high radiation condition. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 32

Event 32 involves a worker falling into the basin. Assuming the activity of the basin water is controlled, the consequences of this event are low. The Basin Water Chemistry Control Program protects this assumption.

Event 334

Event 334 involves the potential for breaching the portable deionizers, associated pipes, hoses, or valves, causing direct shine to the workers. Assuming a worst-case activity for the portable deionizers, it was concluded that this event results in low consequences. The Reactor Deionizer Resin Replacement/Regeneration Program controls the inventory in the portable deionizers. In addition, it is assumed that if this event causes a radiation monitoring alarm, the workers would evacuate the area, thereby mitigating the consequence. The Radiological Protection Program provides radiation monitoring and ensures that trained workers evacuate upon a high radiation condition. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). When conditions have stabilized, the response procedure ensures that the spill is cleaned up in a safe and timely manner.

INJURY TO PERSONNEL - GROUP 5 CRITICALITY

Due to the large amounts of fissile material present in the form of irradiated fuel assemblies and due to the presence of water for moderation, criticality is possible in several areas of the RBOF basins. The postulated criticality can occur through process upset conditions (i.e., internal event initiators) or through NPH-induced conditions. The process-, severe weather-, and earthquake-induced criticalities were analyzed as DBAs and are discussed in detail in Sections 3.4.2.3, 3.4.2.4, and 3.4.2.5, respectively.

Controls for the following events have been selected and documented in Table 5.3.2-5.

Events 24, 34, 35, 36, 37a, 37b, 38, 39, 40, 40a, 40b, 41, 42, 42a, 43, 44, 63, 63a, 64, 65, and 404

Events 24, 34, 35, 36, 37a, 37b, 38, 39, 40, 40a, 40b, 41, 42, 42a, 43, 44, 63, 63a, 64, 65, and 404 involve the potential for postulated criticality in several areas of the RBOF basin. (Events 43, 63, 63a, 64, and 65 are covered in Group 10 below.)

The criticalities involve two major consequence concerns: (1) direct dose from prompt gamma and neutron radiation, and (2) dose from plume immersion (cloud shine and inhalation pathways). A yield of $5.0\text{E}+17$ fissions is bounding for the RBOF process-induced criticality. For the bounding accident and facility receptor dose to the workers, the resulting consequences are moderate.

Event 24

Event 24 postulates a criticality occurring as a dry cask is being immersed in the basin. The consequences to the workers will potentially exceed the moderate consequence threshold. This event is estimated to be beyond extremely unlikely because of the rigorous procedural and design requirements of 10 CFR 71 (Ref. 12) for offsite fuel and the requirement that the onsite cask be received filled with water (unless dry transfer is analyzed to be safe through the Unreviewed Safety Question [USQ] process). The Fuel Receipt and Shipping Program protects these cask requirements.

Event 34

Event 34 postulates a criticality occurring when a fuel container floats and causes a critical mass to accumulate.

The following equipment and Administrative Controls are used to reduce the frequency of criticality:

- The Fuel Receipt and Shipping Program ensures that fuel in cans is weighted such that it has sufficient negative buoyancy.
- The Nuclear Criticality Safety Program directs the canning procedure of fuels at the RBOF and ensures that fuel in cans is weighted such that it has sufficient negative buoyancy.
- The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels.
- The criticality control requirements ensure that fuel is stored properly in accordance with the applicable NSDS.
- The designs of the racks, bundles, buckets, bucket inserts, blocking devices, and hardware provide safe storage of the fuel in accordance with the applicable NSDS.

This event results in moderate consequences to the workers. These consequences are based on maintaining the basin water level (shielding and decontamination factor), shielding, worker evacuation, and the fuel characteristics.

The following equipment and Administrative Controls are used to mitigate criticality:

- Basin water level is maintained by a TSR control and the design feature of the basins and canal system.
- The basin walls provide shielding.
- The fuel characteristics are maintained by the Fuel Receipt and Shipping Program and the cask/fuel receiving requirements.

- Nuclear Incident Monitors (NIMs) alert workers that a criticality with a fission product release is in progress or of the presence of high radiation levels. (The NIM trouble portion of the remote 244-H general alarm and the remote NIM alarm are included for this function.)
- To minimize the consequences, when NIMs alarm, the trained workers are assumed to evacuate.
- The control room wall provides shielding to workers in the control room from a criticality occurring outside the basin or without adequate basin water shielding.
- The Radiological Protection Program provides for monitoring of evacuating workers and performance of dose assessments.
- The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Events 35, 36, 37b, 38, 40, 40a, 41, and 44

Event 35 postulates a criticality occurring due to a fuel storage infraction, causing a critical mass to accumulate. Event 36 postulates a criticality occurring when an external load is dropped on the fuel storage racks, deforming the racks, and causing a critical mass to accumulate. Event 37b postulates a criticality occurring when a fuel rack is snagged by a crane or hoist hook, deforming the racks and causing a critical mass to accumulate. Event 38 postulates a criticality occurring when a fuel bundling infraction is made, causing a critical mass to accumulate. Event 40 postulates a criticality occurring when fuel spills from a basket during movement, causing a critical mass to accumulate. Event 40a postulates a criticality occurring when fuel spills from a shipping cask, causing a critical mass to accumulate. Event 41 postulates a criticality occurring when more than one fuel basket is out of a cask at the same time and brought together, causing a critical mass to accumulate. Event 44 postulates a criticality occurring when disassembled pieces are arranged in an unsafe geometry on the basin floor.

The following equipment and Administrative Controls are used to reduce the frequency of criticality:

- The criticality control requirements direct operators to perform operations involving fuel handling and storage in accordance with the applicable NSDS to prevent a criticality. NSDSs provide operators the criticality limits to follow.
- The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. The program also specifies controls to prevent fuel canning infractions, rack or fuel snagging, bundling infractions, fuel drops and spills, and unauthorized disassembly operations.
- The designs of the racks, bundles, buckets, bucket inserts, blocking devices, and hardware provide safe storage and handling of the fuel in accordance with the

applicable NSDS. Also, the RBOF building design limits the amount of debris that falls into the basin following an external crane incident.

- The 100-ton crane south travel limit switch system and mechanical stop prevent the crane from carrying and dropping a load into a fuel storage basin or from traveling beyond the end of the crane rails. Also, the crane operating procedures direct workers to park the crane in a safe position and to use adequate caution when making lifts.
- The Hoisting and Rigging Program ensures that equipment used for fuel and cask handling is available for use in a condition which minimizes the likelihood of failure.

These events are postulated to result in moderate consequences to the workers. These consequences are based on maintaining the basin water level (shielding and decontamination factor), shielding, worker evacuation, and the fuel characteristics.

The following equipment and Administrative Controls are used to mitigate a criticality:

- Basin water level is maintained by a TSR control and the design feature of the basins and canal system.
- The basin walls provide shielding.
- The fuel characteristics are maintained by the Fuel Receipt and Shipping Program and the cask/fuel receiving requirements.
- Nuclear Incident Monitors (NIMs) alert workers that a criticality with a fission product release is in progress. (The NIM trouble portion of the remote 244-H general alarm and the remote NIM alarm are included for this function.)
- To minimize the consequences, when NIMs alarm, the trained workers are assumed to evacuate.
- The Radiological Protection Program provides for monitoring of evacuating workers and performance of dose assessments.
- The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 37a

Event 37a involves the deformation of a fuel rack resulting in a criticality. This event involves a 100-ton crane load falling in the fuel storage basin due to mechanical failure.

The following equipment and Administrative Controls are used to reduce the frequency of a criticality:

- The 100-ton crane south travel limit switch system and mechanical stop prevent the crane from carrying and dropping a load into a fuel storage basin or from traveling beyond the end of the crane rails.
- The crane operating procedures direct workers to park the crane in a safe position and to use adequate caution when making lifts.
- The Hoisting and Rigging Program ensures that equipment used for fuel and cask handling is available for use in a condition which minimizes the likelihood of failure.
- The criticality control requirements direct operators to perform operations involving fuel handling and storage in accordance with the applicable NSDS to prevent a criticality.
- NSDSs provide operators the criticality limits to follow. The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels.
- The designs of the racks, bundles, buckets, bucket inserts, blocking devices, and hardware provide safe storage and handling of the fuel in accordance with the applicable NSDS.

This event is postulated to result in moderate consequences to the workers. These consequences are based on maintaining the basin water level (shielding and decontamination factor), shielding, worker evacuation, and the fuel characteristics.

The following equipment and Administrative Controls are used to mitigate a criticality:

- Basin water level is maintained by a TSR control and the design feature of the basins and canal system.
- The basin walls provide shielding.
- The fuel characteristics are maintained by the Fuel Receipt and Shipping Program and the cask/fuel receiving requirements.
- Nuclear Incident Monitors (NIMs) alert workers that a criticality with a fission product release is in progress. (The NIM trouble portion of the remote 244-H general alarm and the remote NIM alarm are included for this function.)
- To minimize the consequences, when NIMs alarm, the trained workers are assumed to evacuate.
- The Radiological Protection Program provides for monitoring of evacuating workers and performance of dose assessments.
- The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 39

Event 39 postulates a criticality occurring when fuel debris on the basin floor accumulates into a critical mass. The likelihood of this event is minimized by ensuring fuel stays intact by minimizing corrosion.

The following equipment and Administrative Controls are used to reduce the frequency of a criticality:

- The Basin Water Chemistry Control Program minimizes corrosive materials in the basin water.
- The criticality control requirements ensure that fuel is stored properly in accordance with NSDS requirements. NSDSs provide operators the criticality limits to follow.
- The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. The program also specifies controls to prevent unauthorized disassembly operations.
- The designs of the racks, bundles, buckets, bucket inserts, blocking devices, and hardware provide safe storage and handling of the fuel in accordance with the applicable NSDS.

This event is postulated to result in moderate consequences to the workers. These consequences are based on maintaining the basin water level (shielding and decontamination factor), shielding, worker evacuation, and the fuel characteristics.

The following equipment and Administrative Controls are used to mitigate a criticality:

- Basin water level is maintained by a TSR control and the design feature of the basins and canal system.
- The basin walls provide shielding.
- The fuel characteristics are maintained by the Fuel Receipt and Shipping Program and the cask/fuel receiving requirements.
- Nuclear Incident Monitors (NIMs) alert workers that a criticality with a fission product release is in progress. (The NIM trouble portion of the remote 244-H general alarm and the remote NIM alarm are included for this function.)
- To minimize the consequences, when NIMs alarm, the trained workers are assumed to evacuate.
- The Radiological Protection Program provides for monitoring of evacuating workers and performance of dose assessments.
- The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 40b

Event 40b involves an RRFC (an assay instrument used in the cask basin to measure the U-235 content of the material test reactor fuel assemblies by neutron interrogation from an Am-241/lithium source). This event postulates a criticality occurring in the RRFC due to improper fuel handling around the RRFC or improper loading of the RRFC. It has been determined that with the mechanical barrier (skirt) installed, an acceptable safety margin is provided. Therefore, this event is considered beyond extremely unlikely.

The safety skirt is a design feature and is verified to be installed prior to use of the RRFC. The Fuel Receipt and Shipping Program ensures that fuel characteristics shipped to the RBOF meet the analysis assumptions. The cask/fuel receiving requirements ensure that the unloaded fuel is the fuel described in the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses or is isolated. The criticality control requirements ensure that fuel is stored properly in accordance with NSDS requirements. NSDSs provide operators the criticality limits to follow. The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. Also, the Nuclear Criticality Safety Program specifies controls to prevent fuel drops and spills. The designs of the racks, bundles, buckets, bucket inserts, blocking devices, and hardware provide safe storage and handling of the fuel in accordance with the applicable NSDS.

Event 42

Event 42 postulates a criticality occurring as a result of low water level in the basin, allowing decay heat to deform fuel into a critical mass. The Fuel Receipt and Shipping Program ensures that fuels shipped to the RBOF have a maximum heat load such that fuel does not melt upon low water level when stored in its normal storage configuration. The cask/fuel receiving requirements ensure that the unloaded fuel is the fuel described in the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses or is isolated. The criticality control requirements ensure that fuel is stored properly in accordance with NSDS requirements. NSDSs provide operators the criticality limits to follow. The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. The designs of the racks, bundles, buckets, bucket inserts, blocking devices, and hardware provide safe storage and handling of the fuel in accordance with the applicable NSDS. These controls lower the frequency of a criticality into the beyond extremely unlikely range.

Event 42a

Event 42a postulates a criticality occurring as a result of a fast water leak in the 70-ton shipping cask. A criticality occurring in a shipping cask is estimated to result in a significantly higher dose to the workers than that calculated for a criticality occurring in the basin. The Fuel Receipt and Shipping Program ensures that fuel is loaded in a critically safe arrangement and that the fuel

characteristics meet the analysis assumptions. The 70-ton cask shipping requirements ensure that the cask is properly set up (cadmium plates and blocking devices) in accordance with the applicable NSDS. The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. These controls lower the frequency of a criticality into the beyond extremely unlikely range.

Event 404

Event 404 postulates a criticality occurring as a result of an accumulation of fissile material on the basin purification filter or the deionizers, forming an unsafe geometry. This event is estimated to be beyond extremely unlikely, assuming the basin water activity is controlled and the deionizers have controls for periodic inspection. The Basin Water Chemistry Control Program minimizes fissile materials in the basin water. The Deionizer Resin Replacement/Regeneration Program ensures that fissile resin loading is minimized. The Radiological Protection Program monitors the radiation levels in the building and ensures that workers would be alerted to higher than normal levels on the resin, which would prevent any more fissile material buildup on the resin. These controls lower the frequency of a criticality into the beyond extremely unlikely range.

INJURY TO PERSONNEL - GROUP 6 INTERFACES

The RBOF and RRF contain hazardous energy sources and hazardous material sources capable of causing interface failures during normal routine operations.

Controls for the following events have been selected and documented in Table 5.3.2-6.

Event 51

Event 51 involves the interaction of basin water and offsite fuels containing any material which could result in a violent reaction (e.g., sodium and potassium). Life-threatening injuries from the violent reaction could occur. The other hazardous events involving interfaces result in negligible consequences and require no further discussion for TSR controls. The Fuel Receipt and Shipping Program ensures that the fuel stored does not contain any material (e.g., sodium and potassium) which would react violently with water or has been properly cleaned and canned. The cask/fuel receiving requirements ensure that the unloaded fuel is the fuel described in the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses. Following the violent reaction, the Radiological Protection Program provides radiation monitoring and ensures that trained workers evacuate upon a high radiation condition. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 53

Event 53 involves the failure of radiation monitors and the impact to the workers. Because no work can proceed without proper radiation monitoring, the RCO personnel would alert trained workers to the degraded condition and consequences of the failure of the radiation monitor would be minimal. Therefore, the Radiological Protection Program mitigates this event.

Events 56 and 340

Events 56 and 340 involve fires spreading from nearby facilities to the RBOF and RRF. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). The Fire Protection Program, which involves the response of the fire department, firefighting capabilities, and the limiting of combustibles around the facility, should minimize a fire from spreading from an adjacent facility.

INJURY TO PERSONNEL - GROUP 7 WASTE DISPOSAL

During normal RBOF/RRF operations, both hazardous and nonhazardous waste is produced in the form of solid and liquid wastes.

Controls for the following events have been selected and documented in Table 5.3.2-7.

Events 58 and 342

Events 58 and 342 involve the potential for improper disposal of solid waste. These events do not have the potential for injury to the worker, assuming the workers are qualified and a Waste Management Program and a Radiological Protection Program are established. The Waste Management Program minimizes radioactive materials in the waste through the proper implementation of the WAC for the receiving facility. The Radiological Protection Program ensures that the monitoring occurs during waste handling activities and that the waste handling activities are performed with As Low As Reasonably Achievable (ALARA) goals being achieved. Worker training ensures that the workers follow the Waste Management Program and Radiological Protection Program.

Events 57 and 341

Events 57 and 341 involve the inability to transfer waste from either the cask wash pit or the RRF area into the waste tank. These events have negligible consequences because the Radiological Protection Program ensures that the monitoring occurs during waste handling activities and that the waste handling activities are performed with ALARA goals being

achieved. Also, if the waste is stationary, the monitoring will alert workers to potential increases in radiological consequences so that the trained workers could evacuate if necessary.

INJURY TO PERSONNEL - GROUP 8 MAINTENANCE

Controls for the following events have been selected and documented in Table 5.3.2-8.

Events 59 and 59a

Events 59 and 59a involve routine operations performed in the RBOF and RRF where it is possible for workers to be exposed to varying levels of radiation or contamination. Maintenance activities are carefully planned using ALARA guidelines. It is assumed that all maintenance activities that result in radiation or contamination would be in the worst-case equivalent to the basin contamination, which was annotated low consequences assuming a Radiological Protection Program is established. The Radiological Protection Program provides radiation monitoring during maintenance activities and that the maintenance activities are performed with ALARA goals being achieved.

Event 60

Event 60 involves fire occurring during welding operations. The Fire Protection Program controls welding activities by using a fire watch during welding and by requiring proper safety techniques to be employed. Only workers trained in welding safety and fire watching are allowed to perform welding activities. If a fire did occur, the trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 343

Event 343 involves routine operations performed in the RRF where it is possible for workers to be exposed to varying levels of chemical hazards. Maintenance activities are carefully planned using guidelines established to prevent chemical consequences. These activities are controlled by the Hazardous Material Inventory Control Program and site chemical safety standards. If a release did occur, the trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). Also, workers are trained to attempt to limit the spill size and clean up the spill in a timely fashion; however, this spill response is considered defense in depth.

INJURY TO PERSONNEL - GROUP 9 EXTRINSIC HAZARDS

The effect of extrinsic hazards on the RBOF and RRF can cause a release of chemical inventories and unsafe criticality configurations. Extrinsic hazards include aircraft crash and ground vehicle accident.

Controls for the following events have been selected and documented in Table 5.3.2-9.

Events 345c and 345d

Events 345c and 345d involve a ground vehicle crash resulting in a breach of confinement. Events 345c and 345d result in a moderate consequence and a chemical release exceeding ERPG-2. The events cause damage to the chemical tanks, resulting in spills similar to Events 314a, 316, and 318 with similar consequences. The Hazardous Material Inventory Control Program limits the type (e.g., sodium nitrite, sodium hydroxide, and nitric acid) and amount of chemical inventory available for release. The designs of the chemical tank (e.g., sodium nitrite, nitric acid), chemical pad No. 2 berm, and drain system to waste tank EP38 ensure that sodium nitrite does not mix with nitric acid (see Event 401a). Trained workers follow operating procedures for the chemical pad No. 2 pump, which prevents pumping the contents of chemical pad No. 2 to waste tank EP38 without first sampling for nitric acid. The facility design is such that the chemical tanks are outside the normal traffic pattern. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 345a

Event 345a involves a ground vehicle crashing into a portable deionizer, resulting in a breach of confinement. Assuming a worst-case activity for the portable deionizers, it was concluded that this event results in low consequences. The Reactor Deionizer Resin Replacement/Regeneration Program limits the activity of the resin. In addition, the Radiological Protection Program provides radiation monitoring and ensures that trained workers evacuate upon a high radiation condition. The facility design is such that the portable deionizer is located outside the normal traffic pattern. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). When conditions have stabilized, the response procedure ensures that the spill is cleaned up in a safe and timely manner.

Events 62 and 345

Events 62 and 345 involve aircraft crashes, which were determined to be in the beyond extremely unlikely frequency range based on facility size and frequency of aircraft being over the facility.

INJURY TO PERSONNEL - GROUP 10 NATURAL PHENOMENA

Controls for the following events have been selected and documented in Table 5.3.2-10.

Events 43, 63, 63a, 64, and 65

Events 43, 63, 63a, 64, and 65 discuss criticalities caused by natural phenomena. Event 43 postulates an NPH event that causes a low basin water level, which results in decay heat deforming the fuel and reconfiguring the fuel into a critical mass. Event 63 postulates an NPH event that causes the building structure to fall into the basin, which results in rack deformation and the fuel being reconfigured into a critical mass. Event 63a postulates an NPH event that causes a criticality while the basin is in a draindown condition. Events 64 and 65 evaluate either the building structure or a missile strike that causes fuel to be reconfigured into a critical mass. These events are postulated to result in moderate consequences to the workers.

The Fuel Receipt and Shipping Program ensures that fuels shipped to the RBOF have a maximum heat load such that fuel does not melt upon low water level when in its normal storage configuration. The cask/fuel receiving requirements ensure that the unloaded fuel is the fuel described in the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses or is isolated. The criticality control requirements ensure that fuel is stored properly in accordance with NSDS requirements. NSDSs provide operators the criticality limits to follow. The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. The designs of the racks, bundles, buckets, bucket inserts, blocking devices, and hardware provide safe storage and handling of the fuel in accordance with the applicable NSDS. The basin is postulated to hold water following an earthquake event with only minor leakage. The basin water provide protection to the fuel from missile and debris damage. The 100-ton crane south travel limit switch system and mechanical stop prevent the crane from carrying and dropping a load into a fuel storage basin or from traveling beyond the end of the crane rails. Also, the crane operating procedures direct workers to park the crane in a safe position so that it would not fall into the basin during a building failure. The Hoisting and Rigging Program ensures that equipment used for fuel and cask handling is available for use in a condition which minimizes the likelihood of failure.

If a criticality did occur, these events would result in moderate consequences to the workers. These consequences are based on maintaining the basin water level (shielding and decontamination factor), shielding, worker evacuation, and the fuel characteristics. Before the event, the basin water level is maintained by the design feature of the basins and canal system. After the event the basin and canal system design will be maintained with only minor leakage. Also, the basin walls provide shielding. The fuel characteristics are maintained by the Fuel Receipt and Shipping Program and the cask/fuel receiving requirements. To minimize the consequences, the trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). The Radiological Protection Program provides for monitoring of evacuating workers and

performance of dose assessments. If the basin water does leak from the basin, the Basin Water Chemistry Control Program minimizes the consequences.

Event 347

Event 347 involve the effects of an earthquake or high wind event on the chemical pads. It is anticipated that should an event occur, evacuation of the facilities and area would occur and reentry would be performed as required by applicable event response procedures. The sodium nitrite and nitric acid tanks and chemical pad rain shelters are qualified to survive an evaluation basis earthquake event and prevent a spill. However, no credit for the seismic resistance of these SSCs is assumed (Refer to Chapter 3, Loss of Chemical Dumpster Containment for the SSCs credited). For the high wind events, the potential exist to exceed moderate consequence level thresholds. The consequences of this event can be minimized by limiting the chemicals stored in the tanks, and the trained workers following the applicable event response procedure. The Hazard Material Inventory Control Program limits the chemical inventories. The designs of the nitric acid tank, sodium nitrite tanks, chemical pad No. 2 berm, and drain system to waste tank EP38 ensure that sodium nitrite does not mix with nitric acid. Trained workers follow operating procedures for the chemical pad No. 2 pump, which prevents pumping the contents of chemical pad No. 2 to waste tank EP38 without first sampling for nitric acid. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 347a

Event 347a involves an earthquake or high wind event on the RRF and hose rack. Both the Reactor and the RBOF Deionizer Resin Replacement/Regeneration Program limits the activity of the resin. In addition, the Radiological Protection Program provides radiation monitoring and ensures that trained workers evacuate upon a high radiation condition. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 348

Event 348 postulate severe weather in the form of freezing temperatures, and snow and ice causing a chemical spill. To protect against this type of severe weather, the chemicals are expected to solidify or have a low evaporation rate based on the cold weather or snow and are not expected to have significant consequences. The Hazard Material Inventory Control Program and the tank design limit the chemical inventories. The designs of the chemical tank (e.g., sodium nitrite, nitric acid), chemical pad No. 2 berm, and drain system to waste tank EP38 ensure that sodium nitrite does not mix with nitric acid (see Event 401a). Trained workers follow operating procedures for the chemical pad No. 2 pump, which prevents pumping the contents of chemical pad No. 2 to waste tank EP38 without first sampling for nitric acid. The trained workers follow

the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Event 66

Event 66 involves the rain-induced flood of the basin causing overflow. Assuming the activity of the basin water is controlled this event results in low consequences. The Basin Water Chemistry Control Program limits the activity of the basin water. The building design minimizes the rain that can reach the basin. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.).

Events 67, 69, and 349

Event 67 involves a containment breach following severe weather in the form of snow and ice. The building 244-H design prevents this containment breach. Events 69 and 349 involve lightning causing damage to equipment or the building. Again, the RBOF building design prevents this damage for Event 69. Event 349 results in negligible consequences and requires no prevention or mitigation.

Event 401a

Event 401a involves the generation of nitrous dioxide. This event postulates that nitric acid and sodium nitrite will mix in waste tank EP38. The event could be initiated by an earthquake, high winds, or tornado. It is concluded that this hazard could impact the workers. The sodium nitrite tank, sodium nitrite hold tank, and nitric acid tank are assumed to survive an earthquake event and to remain within the bermed area during a high wind or tornado event. The Hazardous Material Inventory Control Program sets the limit for the sodium nitrite inventory such that it does not overflow the berm. The chemical pad No. 2 berm, and drain system to waste tank EP38 ensure that sodium nitrite does not mix with nitric acid. Also, trained workers follow operating procedures for the chemical pad No. 2 pump, which prevents pumping the contents of chemical pad No. 2 to waste tank EP38 without first sampling for nitric acid. The trained workers follow the applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.). However, the prevention features listed make the frequency of this event in the beyond extremely unlikely range and no mitigation features are required.

5.4 DERIVATION OF FACILITY MODES AND PROCESS AREA DESCRIPTION

5.4.1 MODES

A facility operating mode defines the process objective, the allowable operating parameters, and the allowable maintenance and production activities. In this manner, each mode limits the facility's operational activities, based on their associated hazards, in order to safely and efficiently fulfill the process objective. The modes apply to the process area applicability as described in the LCO.

The mission of the RBOF and RRF is to:

- Receive, handle, and store irradiated nuclear fuel elements from offsite power and research reactors, from foreign country reactors, and from onsite reactors
- Repackage nuclear fuel elements into containers or bundles necessary for extended storage and/or shipment to onsite or offsite facilities
- Provide the provisions to handle, separate, and transfer waste generated from nuclear fuel element storage
- Maintain the RRF to receive resins used in the RBOF and resins shipped from onsite reactor facilities, regenerate the resins, and transfer the waste to the waste tank in the RBOF

The following modes will be established in the TSR:

- Operation Mode is the mode where cask handling is permitted in the RBOF and General Building Area (GBA) (not in the RRF). Fuel handling and fuel storage are permitted in the RBOF (not in the GBA or RRF).
- Standby Mode is the mode where cask handling is permitted in the GBA and RBOF (not in the RRF). Cask handling and movement of loads using the 100-ton crane is not permitted if fuel is present in the cask basin. Fuel storage is permitted in the RBOF (not in the GBA or RRF). Fuel handling is not permitted in the facility. Other activities (e.g., basin water deionizer operation, resin regeneration, chemical transfers) are not restricted during Standby.
- Shutdown Mode is the mode where cask handling is permitted in the GBA (not in the RBOF or RRF). Fuel storage is permitted in the RBOF (not in the GBA or RRF). Fuel handling is not permitted in the facility. Access to the RBOF shall be restricted to only those workers performing activities necessary for responding to alarms, performing surveillances, or restoring LCO compliance (i.e., maintenance on inoperable equipment). Other activities (e.g., basin water deionizer operation, resin regeneration, chemical transfers) are not restricted during Shutdown.

The following definitions were used in determining the facility modes:

- Cask handling shall be the handling or movement of sealed casks (i.e., cask lid secured in place) containing fuel in the facility. Venting is permissible.
- Fuel handling shall be the handling or movement of out-of-cask fuel or unsealed cask containing fuel in the facility and shall include unloading and loading a cask; unsealing and sealing a cask; and transferring, inspecting, disassembling, cropping, handling, bundling, and repackaging fuel in the RBOF building. Fuel handling shall also include the movement of any submerged fuel handling or other grappling device in a basin within two feet of fuel.
- Fuel storage shall be the storage of out-of-cask fuel in its normal storage location and excludes isolated storage. Normal storage is defined in an NSDS.

5.4.2 PROCESS AREAS

The following process areas, as defined in this SAR, may be used in the TSR:

RBOF	Receiving Basin for Offsite Fuels Facility (Building 244-H)
RRF	Resin Regeneration Facility (Building 245-H)
GBA	General Building Area (i.e., the areas outside Buildings 244-H and 245-H, including chemical pads No. 1 and No. 2, storage buildings, office buildings, the yard storage areas, and the surrounding grounds)

5.5 TSR DERIVATION

The review of the RBOF postulated accidents and events discussed in Section 5.3.3.1 indicates that TSR LCOs and Administrative Controls are required for the events discussed in this section. The TSR controls specified in this section focus on preventing or mitigating consequences of the postulated accidents described in Section 5.3.

In addition to the specific TSR controls specified in this section, DOE Order 5480.22 (Ref. 3) provides a discussion of a number of items that are to be included in the Administrative Controls section of the TSR. These items include management responsibilities, specific programs (e.g., quality assurance, configuration control, and environmental compliance), event reporting, and reviews and audits. These standard items do not require facility-specific derivation.

The postulated accidents and their respective required TSR controls are discussed in the following sections.

5.5.1 CHEMICAL SPILL ACCIDENTS

5.5.1.1 Administrative Control: Hazardous Material Inventory Control Program

The Hazardous Material Inventory Control Program sets maximum inventory limits for hazardous materials within the facility. The program shall provide control over new materials brought into the facility and ensure that facility chemical inventories are maintained within the levels assumed in the Authorization Basis. Also, ethylene glycol may not be stored in any location where it would drain into waste tank EP38. Hazardous chemicals are assumed in the hazards analysis not to exceed the established inventory limits. Specific inventory limits require nitric acid to be limited to 1,100 gallons, sodium hydroxide to be limited to 1,100 gallons, and sodium nitrite to be limited to 900 gallons. All other chemicals are less than the reportable quantity, threshold planning quantity, or threshold quantity (unless specifically analyzed through the USQ process).

5.5.1.2 Administrative Control: Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program

The following are some of the accidents and the planned responses relied upon in the accident analysis:

- Spills - alert fellow workers and the shift supervisor; evacuate workers
- Fire - alert fellow workers, the shift supervisor, and the fire department; evacuate workers
- NPHs - evacuate workers

- Radiation alarms - alert RCO; evacuate workers

Many of these responses are proceduralized in abnormal operating and alarm response procedures. Workers are trained to respond to many types of events and when applicable follow these procedures.

Many of the procedures also invoke the Emergency Preparedness Program. As part of the Emergency Preparedness Program, the Area Emergency Plan defines specific measures, policies, and actions to prevent or minimize injuries, damage to property, and impact on the environment caused by accidents, natural disasters, or deliberate damage within the area of responsibility. The Facility Manager, through the Emergency Preparedness Manager, ensures that an Emergency Preparedness Program is established in accordance with applicable DOE requirements, based on formal hazards assessment and requirements. The program contains the following elements:

- Emergency response organization
- Operational emergency event classes
- Notification policies and procedures
- Assessment of the consequences
- Protective actions
- Medical support
- Recovery and re-entry procedures
- Emergency facility and equipment
- Training
- Drills
- Program administration

The SRS Emergency Preparedness Plan defines actions that provide protection for the workers during emergency conditions. The plan provides pre-planned responses and actions that are designed to provide the proper response to emergency conditions that have been reviewed. The Facility Manager has the option to use the Emergency Preparedness Program to move workers to a safe area, protecting all onsite receptors both inside and outside the facility.

5.5.1.3 Administrative Control: Worker Training and Qualification Program

The Worker Training and Qualification Program is in place to ensure that workers are properly trained to perform their duties. Training is provided by the Operations Training Section. The Worker Training and Qualification Program is applicable for normal operations and in accident situations that require worker response. It also provides support to other programs and ensures that credited programs and SSCs have adequate worker training supporting operation and maintenance. One such program is the Radiological Protection Program, which assumes that

workers will respond properly following radiation alarms. Another program is the Emergency Preparedness Program, which assumes that workers will respond properly during emergency conditions.

5.5.1.4 Administrative Control: Chemical Pad No. 2 Berm Pumping Procedures

This Administrative Control ensures that a spill of sodium nitrite will not be pumped into waste tank EP38 when the tank contains nitric acid. This Administrative Control was assumed in the analysis to prevent sodium nitrite from mixing with nitric acid.

5.5.1.5 Design Feature: Chemical Pad No. 2 Berm and Drain System to Waste Tank EP38

These design features ensure that a spill of sodium nitrite will remain within the bermed area. The chemical pad No. 2 berm, and drain system were assumed in the analysis to prevent sodium nitrite from mixing with nitric acid.

5.5.2 SURFACE VEHICLE IMPACT ACCIDENT

5.5.2.1 Design Feature: Facility Design

This design feature ensures that chemical pads, portable deionizers, and hose boxes are out of the normal traffic pattern at RBOF. The facility design was assumed in the analysis to reduce the probability of vehicles reaching the chemical pads, portable deionizers, and hose boxes.

5.5.2.2 Design Feature: Nitric Acid and Sodium Nitrite Chemical Tanks

These design features ensure that the inventories of the chemicals do not mix by remaining within their respective chemical pad berm during NPH events.

The following items discussed previously also reduce the frequency and mitigate surface vehicle impact accidents:

- Hazardous Material Inventory Control Program (discussed in Section 5.5.1.1)
- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Chemical pad No. 2 berm pumping procedures (discussed in Section 5.5.1.4)

- Chemical Pad No. 2 Berm, and Drain System to Waste Tank EP38 (discussed in Section 5.5.1.5)

5.5.3 PROCESS-INDUCED CRITICALITY

5.5.3.1 Administrative Control: Fuel Receipt and Shipping Program

The Fuel Receipt and Shipping Program reduces the frequency of criticality hazards and the release of radionuclides. The consequences of analyzed accidents are based on the assumption that fuel assemblies received meet the fuel characteristics required by the Fuel Receipt and Shipping Program.

Procedures governing fuel receipt and shipping are consistent with applicable DOE and DOT requirements governing fuel receipt and shipping. These procedures are approved, maintained, and adhered to for all related operations. The Fuel Receipt and Shipping Program ensures that fuel received by the RBOF meets the Authorization Basis assumptions regarding the following:

- Isotopic characteristics
- Minimum cooling time
- Maximum heat load
- Maximum radionuclide inventory for an individual fuel unit
- Maximum facility radionuclide inventory

All fuel shipments are screened in accordance with the USQ process. Fuel shipments not meeting these Authorization Basis assumptions are evaluated under the USQ process prior to receipt by the RBOF. Chapter 3 indicates that all of the analyzed accidents have consequences that are below the EGs for the maximally exposed offsite individual; these results are based on the assumption that fuel assemblies received meet the fuel characteristics required by the Fuel Receipt and Shipping Program.

The Fuel Receipt and Shipping Program also ensures the following:

- The casks have documentation (offsite: Safety Analysis Report for Packaging and Certificate of Compliance, onsite: WSRC Procedure Manual 19Q [Ref. 11]).
- Casks are received with fuel loaded in a critically safe arrangement.
- The containment design of onsite casks is maintained, and casks (offsite or onsite) containing flammable antifreeze are not purged to waste tank EP38.
- Casks shipped into and out of the facility are not a deflagration concern.
- Fuel received by the RBOF meets the Authorization Basis assumptions regarding the absence of sodium and potassium.

- The fuel weight/storage requirements used in the structural analysis are controlled.
- The fuel containers are weighted such that they have sufficient negative buoyancy.
- The number of cask receipts, fuel movements, baskets unloaded, etc., is maintained within the assumptions of the frequency analyses.
- Casks are received with water unless determined to be safe dry.
- Minimum water depth above active fuel.

5.5.3.2 Limiting Condition for Operation: Cask/Fuel Receiving Requirements

The cask/fuel receiving requirements are required to reduce the frequency and mitigate a criticality and to ensure that the fuel characteristics received at the RBOF is within the RFA characteristics. Implicit to this LCO is the proper performance by workers trained in accordance with the Worker Training and Qualification Program. The requirement of this LCO is that fuel identification shall match the cask shipping paperwork (loading diagram, DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses. Fuel identification is verified prior to storing fuel in a manner other than isolated.

5.5.3.3 Administrative Control: Nuclear Criticality Safety Program

The Nuclear Criticality Safety Program controls are required to prevent a criticality. Implicit to this administrative control is the proper performance by workers trained in accordance with the Worker Training and Qualification Program. The following are some of the aspects of the program requiring trained workers:

- Traveling bridge operation (rack or fuel snagging)
- Handling of fuel elements, assemblies, bundles, buckets, etc.
- Fuel cropping, bundling, or canning operations
- Movement of loaded casks and cask baskets
- Fuel disassembly operations determined to be safe

The Nuclear Criticality Safety Program requires analyses for all types of fuel or new operations. For fuel types, requirements from these analyses are captured in NSDSs. NSDSs are used to document the results of calculations to determine that criticality safety is ensured. Each fuel shipment is evaluated prior to storage under the Fuel Receipt and Shipping Program. If the shipment meets the requirements of the program, approved storage location and proper spacing requirements for safe storage are determined prior to shipping. Programmatic controls have been implemented to analyze all proposed storage locations for fuel to ensure criticality safety. These safety requirements derived from the analyses are documented in the NSDS and specify the following applicable requirements:

- The normal storage location
- The maximum safe number of fuel pieces (plates, assemblies, disassembled pieces, bundles, etc., as defined in the NSDS) that may be safely stored together in a safe fuel unit
- The maximum number of safe fuel units that may be safely stored in a safe fuel group
- The required safe spacing between adjacent rows of safe fuel groups of the same fuel type and of differing fuel types
- SRS 70-ton cask shipping requirements

Verification that an NSDS applicable to the fuel type being handled has been prepared and approved prior to opening the applicable shipping cask and prior to fuel handling is performed.

5.5.3.4 Limiting Condition for Operation: Criticality Control Requirements

The criticality control requirements are required to prevent a criticality. Implicit to this LCO is the proper performance by workers trained in accordance with the Worker Training and Qualification Program. The following are the requirements and SRs of the LCO requiring trained workers:

- Fuel shall be stored in accordance with an approved NSDS in the #1 or #2 storage basin or shall be isolated. This requirement is checked during and after fuel movements.
- Cask baskets containing fuel pieces numbering greater than the maximum safe number that may be stored in a safe fuel unit shall not be removed from a cask until removal of the cask basket has been determined to be safe by the USQ process. This requirement is checked prior to movement of fuel in cask baskets.
- Significant quantities of materials that are composed of appreciably better neutron reflections than water, but are not part of a safe fuel group, shall be kept at least 12 inches from fuel unless the materials are necessary for operations and the presence of the material has been determined to be safe by the USQ process. This requirement is checked once every 7 days.
- Support grating for fuel in an upper tier shall be installed at its NSDS required spacing above the lower tier prior to fuel being moved onto the upper tier. This requirement is checked prior to moving fuel requiring grating for separation.

The SRs were chosen to ensure that criticality safety is maintained in all operations.

5.5.3.5 Limiting Condition for Operation: SRS 70-Ton Cask Shipping Requirements

The SRS 70-ton cask shipping requirements are required to prevent a criticality. Implicit to this LCO is the proper performance by workers trained in accordance with the Worker Training and Qualification Program. The following are the requirements and SRs of the LCO requiring trained workers:

- The 70-ton is configured as required by the applicable fuel's NSDS.
- The 70-ton cask cadmium plates reduce neutron transmission to $\leq 50\%$. This requirement is checked through performance of a neutron transmission verification within 12 months prior to loading a cask with fuel.

5.5.3.6 Limiting Condition for Operation: 100-Ton Crane Rail South Travel Limit Switch System

The crane limit switch system stops the crane movement, preventing the crane from traveling south over the #2 storage basin and contacting the mechanical stops, thus preventing the 100-ton crane from dropping a load into the #2 storage basin, which could initiate a criticality. When actuated by the stop bar, the limit switch system provides braking power to the crane motor. To ensure operability of the limit switch system, an LCO is required.

This LCO shall require the 100-ton crane to be positioned north of the #2 storage basin during Operation, Standby, and Shutdown Modes and ensure operability of the limit switch system.

Procedures direct workers to park the crane in a safe position such that it will not initiate a criticality. The use of the crane during movement near the basin or with casks is considered a "lift."

SRs shall be established to ensure the following:

- The crane is positioned north of the #2 storage basin. This surveillance shall be performed every 24 hours during Operation and Standby, prior to entering Shutdown, and every 7 days during Shutdown. The 24-hour frequency during Operation and Standby is based on the assumption that the crane limit switch system is operable.
- An instrument loop test shall be performed on the 100-ton-capacity crane stop switch every 3 months. This surveillance frequency is based on engineering judgment and operating experience.
- The 100-ton-capacity crane stop bar will actuate the crane limit switch. This surveillance shall be performed every 12 months. This frequency is based on engineering judgment and operating experience.

This LCO requires no cross-references to other LCOs.

5.5.3.7 Design Feature: Crane Rail South Mechanical Stops

This design feature ensures that the crane does not travel past the end of the rail and fall into a basin containing fuel. This design feature was assumed to minimize a criticality from the crane falling in the basin and reconfiguring fuel.

5.5.3.8 Administrative Control: Hoisting and Rigging Program

This Program ensures that the hoisting and rigging equipment is sufficiently maintained and tested so that it performs its intended function. The equipment was assumed to perform its intended function with a greater reliability than human failures.

5.5.3.9 Limiting Condition for Operation: Nuclear Incident Monitoring System

This LCO shall be established to provide indication to the workers of a nuclear incident (direct radiation or fission product gas). This LCO shall require three NIM pairs to be operable. Since the depth of water over the fuel is less than 10 feet (about 4 feet 6 inches in some cases), the NIMs may alarm from either direct radiation from the criticality or fission product gases released from the fuel and basin water surface. The NIMs may also alarm as a result of direct radiation unrelated to criticality.

The NIMs consist of pairs of NIM units and their associated local alarms. The NIM units are self-contained instruments. Each unit contains an ion chamber, indicating meter, battery, alarm bell, and associated electronics. When incident gamma radiation causes a pre-set radiation rate level to be exceeded, the alarm bells sound. For redundancy purposes, NIM units are mounted in pairs at each selected monitoring site so that if one unit of a pair malfunctions, the other unit maintains coverage for the associated coverage area. Electrical power for NIMs comes directly from the RBOF building electrical system. Each NIM is attached to a battery reserve, enabling it to respond during a power failure. The NIM systems are designed to detect gamma radiation and to provide audible and visual alarms at a radiation rate between 0.5 and 3 R/hr. The alarm bell warns workers to evacuate the vicinity in the event of a nuclear incident that could pose a hazard to workers. Remote alarm bells warn workers in the vicinity of RBOF.

Initial testing and inspection of the area verifies that the NIMs are made in accordance with specifications. System response to radiation is measured quarterly to confirm continuing instrument performance, and the system is recalibrated at least annually. Records of tests and recalibration are maintained. The entire area alarm system is tested annually. Each bell circuit is tested at least once every 3 months. Field tests verify that the signal is audible above background noise. A measurement of alarm decibel levels is performed at least once every 36 months following an original baseline measurement.

Chapters 2 and 6 provide a detailed discussion of NIMs in the RBOF.

SRs shall be established to perform the following:

- Verify that three RBOF NIM pairs are in service on a monthly basis, every 12 hours during Operation and Standby, and every 112 hours during Shutdown
- Perform an instrument loop check on each of the six RBOF NIM units on a 3-month basis
- Perform an instrument loop test on each of the six RBOF NIM units on a yearly basis
- Perform an instrument loop calibration on each of the six RBOF NIM units on a yearly basis

This LCO requires cross-references to the LCOs for the remote 244-H general alarm and the remote NIM alarm.

5.5.3.10 Limiting Condition for Operation: Remote 244-H General Alarm

This LCO shall be established to provide indication of a NIM trouble alarm to the command function personnel in the Building 221-H Control Room during an RBOF facility shutdown. The remote 244-H general alarm is credited in the accident analyses as a means of monitoring the operability of the NIMs during facility shutdown. NIM inoperability is a condition outside of normal operations and would require entry into the Actions section of the NIM LCO.

SRs shall be established to perform the following:

- Verify that the remote 244-H general alarm is operable within 4 hours prior to entering Shutdown
- Perform an instrument loop check on the remote 244-H general alarm on a 3-month basis

This LCO requires cross-references to the LCOs for the NIMs and the remote NIM alarm.

5.5.3.11 Limiting Condition for Operation: Remote NIM Alarm

This LCO shall be established to provide indication of a nuclear incident (direct radiation or fission product gas) to the command function personnel in the Building 221-H Control Room during an RBOF facility shutdown. The remote NIM alarm is credited in the accident analyses as a means of alerting facility workers to a nuclear incident or high radiation levels. The remote NIM alarm in the Building 221-H Control Room would provide an indication to site workers of a potential nuclear incident or high radiation level in the RBOF during Shutdown.

SRs shall be established to perform the following:

- Verify that the remote NIM alarm is operable within 4 hours prior to entering Shutdown and every 12 hours
- Perform an instrument loop test on the remote NIM alarm on a 3-month basis

This LCO requires cross-references to the LCOs for the remote 244-H general alarm and the NIMs.

5.5.3.12 Limiting Condition for Operation: Basin Water Level

The basin water level surveillance ensures that the minimum water level needed is present to provide shielding and the decontamination factor assumed in the analysis is present at all times.

An LCO shall be established to ensure that basin water level is maintained at 4 feet 6 inches over the top of all stored fuel assemblies. Adequate basin water level is assumed in the SAR as an initial condition of waterborne accidents. Basin water is credited in the consequence analysis of a criticality as an attenuation medium for reducing prompt worker exposure and as a decontamination medium for reducing consequences to both the workers and public. The basin water also provides secondary or tertiary containment for fission products contained in the stored fuel. Therefore, a minimum required basin level is included in an LCO. An additional analysis was performed showing that the prompt dose to the workers standing on the RBOF traveling bridge (4 inches above the floor) is less than 100 rem with at least 4 feet 6 inches of water over the fission source. The basin water also provides protection of the fuel from missiles and debris that may fall into the basin and damage the fuel.

Four feet six inches of water over the highest basin fission source corresponds to -19 inches with respect to the RBOF floor grade. To allow for conservatism, 1 inch was added to the basin level to give -18 inches. Maintaining the RBOF basin water level higher than -18 inches with respect to the RBOF floor grade would therefore be sufficient shielding and decontamination factor to ensure that workers do not receive a prompt fatal dose as the result of a criticality. In addition, the decontamination factors ensure that the public does not receive a dose greater than the EGs.

Fuel in transit during fuel handling is not postulated to initiate a criticality, unless dropped to the level of stored fuel into a critical configuration, and thus does not require a minimum water coverage. Water coverage is required for normal radiation protection during fuel handling. In the event that adequate water level is lost, operable NIMs would serve as a secondary line of defense for worker protection in the event of a criticality.

An LCO shall require the RBOF basin water level to be higher than -18 inches with respect to the RBOF floor grade. This limit applies to any isolated basin containing fuel. If the basins are connected and water level is uniform, representative measurement of one basin is sufficient.

The LCO limit was chosen conservatively to provide an additional margin of safety. Measurement uncertainties were not factored into the values chosen. The values represent the values as stated in Chapter 3. This LCO does not require any active basin monitoring

instrumentation (local or remote) to be operable. Local water level indication will provide sufficient water level monitoring capability for the purposes of this LCO.

SRs shall be established to ensure that the basin water level is higher than -18 inches with respect to the RBOF floor grade. Monitoring the basin water level ensures that when minimum level requirements are not met, appropriate actions are taken in a timely manner to minimize the radiological worker consequences. The 12-hour frequency during Operation and Standby is based on the need for detection of the decrease in basin water level and is substantiated through the low probability of rapid change in basin water level. The 112-hour frequency during shutdown is required to detect water loss when access to the RBOF is limited and is based on the frequency analysis of Chapter 3. Based on this frequency, it was assumed that a basin draindown coupled with an NPH-induced criticality would be beyond extremely unlikely.

5.5.3.13 Administrative Control: Basin Water Chemistry Control Program

The Basin Water Chemistry Control Program has been established to limit the basin water activity and to minimize the potential for corrosion in the RBOF basins. The program directs the use of operating procedures, including remedial actions to be taken whenever the program limits are exceeded. The SAR analysis assumes that the Basin Water Chemistry Control Program is in effect. Under the program, basin water is monitored, sampled, and analyzed, and limitations are placed on the basin water activity, basin water pH, basin water conductivity, basin water temperature, and basin water copper and mercury content. The following are the limits on the basin water (Ref. BW Chemistry Report - see Chapter 2):

- Basin water activity $< 7\text{E-}03 \text{ } \mu\text{Ci/ml}$
- Basin water pH $5.5 \leq \text{pH} \leq 8.5$
- Basin water conductivity $< 10 \text{ } \mu\text{mho/cm}$
- Basin water temperature $\leq 40^\circ\text{C}$
- Basin water copper $\leq 0.1 \text{ ppm}$
- Basin water mercury $\leq 0.1 \text{ ppm}$
- Basin water chloride $\leq 0.1 \text{ ppm}$
- Basin water iron $\leq 2.0 \text{ ppm}$
- Basin water aluminum $\leq 2.0 \text{ ppm}$

The Basin Water Chemistry Control Program limits on basin water activity (activity $< 7\text{E-}03 \text{ } \mu\text{Ci/ml}$) shall ensure that the consequences of a basin water release will not exceed the analysis. In addition, this activity limit ensures that the alpha emitters (assumed to be fissionable materials) remain below the assumed amount analyzed to be collected on the deionizer system resins. Therefore, a criticality occurring in the deionizer was assumed to be beyond extremely

unlikely. Also, by limiting the alpha emitters on the resin, self-ignition of the resin was discounted as an ignition source.

5.5.3.14 Administrative Control: Radiological Protection Program

The Radiological Protection Program ensures that procedures for worker radiation protection are prepared consistent with DOE requirements and are approved, maintained, and adhered to for all operations involving worker radiation exposure. Radiation monitoring shall be in place during cask movement during fuel handling activities, and before re-entry into the RBOF when the facility is unmanned, to provide advance warning to the workers in the event of an inadvertent exposure, such as when fuel is uncovered.

The Radiological Protection Program shall ensure that the radiation exposure of workers is maintained within applicable DOE limits and ALARA principles. This program serves to protect all workers both inside and outside the facility. Also, radiation monitoring shall ensure that, if a process criticality occurs, the workers will receive adequate warning to permit evacuation before the workers are exposed to a level of radiation that results in serious injury or death. The program ensures monitoring of evacuating workers and performance of dose assessment following radiological events. The program provides radiation monitoring for areas of potentially high radiation. Included in the Radiological Protection Program are requirements to monitor particulate air filters which ensures that the 100-Ci Cs-137 inventory assumption of the accident analysis is maintained. The SAR analyses assume that these aspects of the Radiological Protection Program are in effect. Proper performance and response by workers, who are trained in accordance with the Worker Training and Qualification Program, are implicit to the Radiological Protection Program.

5.5.3.15 Administrative Control: Deionizer Resin Replacement/Regeneration Program

The Deionizer Resin Replacement/Regeneration Program is in place to ensure that the RBOF deionizer bed resin is periodically monitored, replaced, or regenerated consistent with the radioisotopic buildup assumptions of the hazards and accident analysis. The analyses discussed in Sections 3.3 and 3.4 assumed that the maximum curie content of the deionizer was 1,060 Ci distributed over isotopes given in Tables 3.4-10 and 3.4-12. In addition, the criticality analysis assumed that before sufficient fissionable material would accumulate on the deionizer resins, the resins would be replaced. Also, by controlling the amount of alpha emitters on the resins, the potential for self-ignition and criticalities is diminished. This program considers the types of radioactive material available for resin deposit, potential rates of accumulation, and monitoring methods.

5.5.3.16 Design Feature: Basins and Canal System

This design feature ensures that the basin water is confined in the basins and canals. The basins and canal system also provides shielding to workers outside the RBOF building from a criticality. The basins and canals are assumed to survive an earthquake with only minor leakage occurring from the basin.

5.5.3.17 Design Feature: Storage Racks, Bundles, Buckets, Bucket Inserts, Blocking Devices, and Associated Hardware

This design feature ensures that this equipment maintains fuel in critically safe spacing during movement and storage and to prevent fuel melting during a reduced water level condition. The equipment was assumed to perform its intended safety function during normal operations, and earthquakes. For high wind events, the storage racks, etc., reduce the probability of a criticality.

5.5.3.18 Design Feature: Control Room Concrete Wall

This design feature provides shielding to workers in the control room during a criticality that occurs outside the basin. This design feature was assumed to help mitigate a criticality.

5.5.3.19 Design Feature: RRFC Safety Skirt

This design feature ensures that fuel assemblies (all dimensions greater than 2 inches) cannot slide underneath the bottom plate of the RRFC. This design feature was assumed to help reduce the frequency of a criticality.

5.5.3.20 Design Feature: SRS 70-Ton Cask

This design feature ensures containment of materials in the cask. This design feature was assumed to help reduce the frequency of a criticality and leaks.

Along with the other Administrative Controls, design features, and LCOs, the following items discussed previously help mitigate the criticality event:

- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)

5.5.4 SEVERE WEATHER

5.5.4.1 Chemical Spills (Induced by Freezing Temperatures, Earthquakes, and High Winds)

The chemical spill accidents caused by severe weather lead to a breach of the chemical dumpsters. The following items reduce the frequency and mitigate severe weather chemical spills accidents:

- Hazardous Material Inventory Control Program (discussed in Section 5.5.1.1)
- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Chemical pad No. 2 berm pumping procedures (discussed in Section 5.5.1.4)
- Chemical Pad No. 2 Berm, and Drain System to Waste Tank EP38 (discussed in Section 5.5.1.5)
- Nitric Acid and Sodium Nitrite Chemical Tanks (discussed in Section 5.5.2.2)

5.5.4.2 High Wind-Induced Criticality

DESIGN FEATURE: 244-H BUILDING

This design feature ensures that the 244-H building withstands severe weather according to the fragility curve used in the frequency analyses. The 244-H building were assumed to withstand certain levels of high winds, earthquakes, snow, ice, rain, and lightning. Also, because of the relatively small size, meteorites and aircraft striking the 244-H building was determined to be beyond extremely unlikely.

In addition to the 244-H building, the following items help reduce the frequency and mitigate a high wind-induced criticality:

- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Fuel Receipt and Shipping Program (discussed in Section 5.5.3.1)
- Cask/fuel receiving requirements (discussed in Section 5.5.3.2)
- Nuclear Criticality Safety Program (discussed in Section 5.5.3.3)

- Criticality control requirements (discussed in Section 5.5.3.4)
- 100-ton crane rail south travel limit switch system (discussed in Section 5.5.3.6)
- Crane rail south mechanical stops (discussed in Section 5.5.3.7)
- Hoisting and Rigging Program (discussed in Section 5.5.3.8)
- Basin water level (discussed in Section 5.5.3.12)
- Radiological Protection Program (discussed in Section 5.5.3.14)
- Basin walls - shielding (discussed in Section 5.5.3.16)
- Basins and canal system (discussed in Section 5.5.3.16)
- Storage racks, bundles, buckets, bucket inserts, blocking devices, and associated hardware (discussed in Section 5.5.3.17)
- Control Room walls - shielding (discussed in Section 5.5.3.18)

5.5.4.3 Fuel Breach (Induced by Earthquakes and High Winds)

The following items help reduce the frequency and mitigate a fuel breach induced by an earthquake or high winds:

- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Fuel Receipt and Shipping Program (discussed in Section 5.5.3.1)
- Cask/fuel receiving requirements (discussed in Section 5.5.3.2)
- Nuclear Criticality Safety Program (discussed in Section 5.5.3.3)
- Criticality control requirements (discussed in Section 5.5.3.4)
- 100-ton crane rail south travel limit switch system (discussed in Section 5.5.3.6)
- Crane rail south mechanical stops (discussed in Section 5.5.3.7)
- Hoisting and Rigging Program (discussed in Section 5.5.3.8)
- Basin water level (discussed in Section 5.5.3.12)
- Basins and canal system (discussed in Section 5.5.3.16)
- 244-H building design (discussed in Section 5.5.4.2)

5.5.5 EARTHQUAKE

5.5.5.1 NPH-Induced Criticality

The following items help reduce the frequency and mitigate an earthquake-induced criticality:

- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Fuel Receipt and Shipping Program (discussed in Section 5.5.3.1)
- Cask/fuel receiving requirements (discussed in Section 5.5.3.2)
- Nuclear Criticality Safety Program (discussed in Section 5.5.3.3)
- Criticality control requirements (discussed in Section 5.5.3.4)
- 100-ton crane rail south travel limit switch system (discussed in Section 5.5.3.6)
- Crane rail south mechanical stops (discussed in Section 5.5.3.7)
- Hoisting and Rigging Program (discussed in Section 5.5.3.8)
- Basin water level (discussed in Section 5.5.3.12)
- Basin Water Chemistry Control Program (discussed in Section 5.5.3.13)
- Radiological Protection Program (discussed in Section 5.5.3.14)
- Basins and canal system (discussed in Section 5.5.3.16)
- Storage racks, bundles, buckets, bucket inserts, blocking devices, and associated hardware (discussed in Section 5.5.3.17)
- 244-H building design (discussed in Section 5.5.4.2)

5.5.5.2 Breach of Pool Basin

The frequency of this event has been determined to be beyond extremely unlikely on the demonstrated basis that the basins and canal system (discussed in Section 5.5.3.16) withstand the Performance Category 3 earthquake with minimal leakage. For the small amount of basin water that does leak, the Basin Water Chemistry Control Program (discussed in Section 5.5.3.13) ensures that the consequences are low. Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program ensures that workers respond properly to event conditions (discussed in Section 5.5.1.2).

5.5.5.3 Breach of Waste Tank EP38

The following programs help reduce the frequency and mitigate a breach of waste tank EP38:

- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Radiological Protection Program (discussed in Section 5.5.3.14)
- Deionizer Resin Replacement/Regeneration Program (discussed in Section 5.5.3.15)

5.5.5.4 Loss of Chemical Dumpster Containment

The chemical spill accidents caused by earthquakes lead to a breach of the chemical dumpsters. The following items reduce the frequency and mitigate earthquake-induced chemical spills accidents:

- Hazardous Material Inventory Control Program (discussed in Section 5.5.1.1)
- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Chemical pad No. 2 berm pumping procedures (discussed in Section 5.5.1.4)
- Chemical Pad No. 2 Berm, and Drain System to Waste Tank EP38 (discussed in Section 5.5.1.5)
- Nitric Acid and Sodium Nitrite Chemical Tanks (discussed in Section 5.5.2.2)

5.5.6 RESIN FIRE

5.5.6.1 Reactor Deionizer Resin Replacement/Regeneration Program

The Reactor Deionizer Resin Replacement/Regeneration Program is in place to ensure that the deionizer bed resin transported from the Reactor Areas is periodically monitored, replaced, or regenerated consistent with the radioisotopic buildup assumptions of the hazards and accident analysis. The analyses discussed in Sections 3.3 and 3.4 assumed that the maximum curie content of the deionizer was 5 Ci, which was conservatively increased to 50 Ci distributed over isotopes given in Table 3.4-13. In addition, the criticality analysis assumed that before sufficient fissionable material would accumulate on the deionizer resins, the resins would be replaced.

Also, by controlling the amount of alpha emitters on the resins, the potential for self-ignition is diminished. This program considers the types of radioactive material available for resin deposit, potential rates of accumulation, and monitoring methods.

In addition to the Administrative Controls, the following programs help reduce the frequency and mitigate a resin fire:

- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Radiological Protection Program (discussed in Section 5.5.3.15)

5.5.7 EXPLOSION

The following programs help reduce the frequency and mitigate explosions:

- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Basin Water Chemistry Control Program (discussed in Section 5.5.3.14)
- Radiological Protection Program (discussed in Section 5.5.3.15)
- Deionizer Resin Replacement/Regeneration Program (discussed in Section 5.5.3.16)
- Reactor Deionizer Resin Replacement/Regeneration Program (discussed in Section 5.5.6.1)

5.5.8 UNCONTROLLED CHEMICAL REACTION

The following items help reduce the frequency and mitigate uncontrolled chemical reactions:

- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Fuel Receipt and Shipping Program (discussed in Section 5.5.3.1)
- Cask/fuel receiving requirements (discussed in Section 5.5.3.2)
- Radiological Protection Program (discussed in Section 5.5.3.14)

5.5.9 AIRCRAFT IMPACT

The frequency of this event has been determined to be beyond extremely unlikely based on the building design discussed in Section 5.5.4.2.

5.5.10 GROUP 1: FIRE/EXPLOSIONS

5.5.10.1 Administrative Control: Fire Protection Program

The Fire Protection Program is designated as safety significant. In general, fire within the RBOF complex will pose no undue risk to the public or workers. The facility is built with a noncombustible construction. There is a lack of continuity of combustibles in most of the RBOF and RRF areas. Also, the nature of the operation within the facility does not generally involve high temperatures or combustion. No other facilities are close enough to the RBOF and RRF to impact this area if fires or explosions should occur in adjacent areas. These conclusions are based on the Fire Hazards Analysis for the RBOF and RRF (Ref. 13) discussed in Chapter 3. The Fire Hazards Analysis evaluates the fire protection and life safety features of the facilities against the fire safety objectives of DOE Order 5480.7A (Ref. 14).

The Fire Protection Program incorporates features of fire prevention and fire response. The preventive features include limiting flammable materials within the RBOF (including limiting the time the tractor which delivers casks is in the carport) and training workers. The fire response features include the fire response teams, site fire departments, worker training in the use of fire extinguishers, and response to fire alarms. The Fire Protection Program provides controls to ensure that appropriate fire protection measures (including welding and compressed gas safety) are maintained to protect the facility from fire and to ensure the capability to maintain the facility in a safe condition in the event of a fire.

In addition to the Fire Protection Program, the following items help reduce the frequency and mitigate fires and explosions:

- Hazardous Material Inventory Control Program (discussed in Section 5.5.1.1)
- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Fuel Receipt and Shipping Program (discussed in Section 5.5.3.1)
- Cask/fuel receiving requirements (discussed in Section 5.5.3.2)
- Basin Water Chemistry Control Program (discussed in Section 5.5.3.13)
- Radiological Protection Program (discussed in Section 5.5.3.14)
- Deionizer Resin Replacement/Regeneration Program (discussed in Section 5.5.3.15)

- SRS 70-Ton Cask Design (discussed in Section 5.5.3.20)
- Reactor Deionizer Resin Replacement/Regeneration Program (discussed in Section 5.5.6.1)

5.5.11 GROUP 2: LOSS OF CONTAINMENT

5.5.11.1 Administrative Control: Waste Management Program

The Waste Management Program ensures that the contents of all waste that leaves the RBOF and RRF meets the WAC of the receiving facility.

5.5.11.2 Administrative Control: Configuration Control Program for Transfer Lines

The configuration of the transfer lines used by RBOF to the tank farm ensures that transfer lines leak underground, mitigating the worker effects.

5.5.11.3 Administrative Control: Reactor Basin Water Chemistry Control Program

The Reactor Basin Water Chemistry Control Program has been established to limit the basin water activity and to minimize the potential for corrosion in the reactor basins. The program directs the use of operating procedures, including remedial actions to be taken whenever the program limits are exceeded. The RBOF SAR analysis assumes that the Reactor Basin Water Chemistry Control Program is in effect to maintain the basin water activity within the consequence analyses.

In addition to the Administrative Controls, the following items reduce the frequency or mitigate the loss-of-confinement events:

- Hazardous Material Inventory Control Program (discussed in Section 5.5.1.1)
- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Fuel Receipt and Shipping Program (discussed in Section 5.5.3.1)
- Cask/fuel receiving requirements (discussed in Section 5.5.3.2)
- Nuclear Criticality Safety Program (discussed in Section 5.5.3.3)
- Criticality control requirements (discussed in Section 5.5.3.4)
- Cask Design (discussed in Section 5.5.3.20)

- Reactor Waste Management Program (discussed in Section 5.5.11.14)
- Hoisting and Rigging Program (discussed in Section 5.5.3.8)
- Nuclear incident monitoring system (discussed in Section 5.5.3.9)
- Remote 244-H general alarm (discussed in Section 5.5.3.10)
- Remote NIM alarm (discussed in Section 5.5.3.11)
- Basin water level (discussed in Section 5.5.3.12)
- Basin Water Chemistry Control Program (discussed in Section 5.5.3.13)
- Radiological Protection Program (discussed in Section 5.5.3.14)
- Basins and canal system (discussed in Section 5.5.3.15). The discharge piping of the filter deionizer basin water pumps are limited in depth to prevent a complete draindown of the basin. This design feature was assumed to prevent draining of the basin with a discharge piping failure.
- RBOF Deionizer Resin Replacement/Regeneration Program (discussed in Section 5.5.3.15)
- Reactor Deionizer Resin Replacement/Regeneration Program (discussed in Section 5.5.6.1)

5.5.12 GROUP 3: LOSS OF CONFINEMENT

5.5.12.1 Administrative Control: Site Safety Program

The Site Safety Program ensures that the workers conduct all work onsite in a safe manner. In particular, the vehicle safety portion of using a spotter when backing up a trailer was assumed in the analyses.

In addition to the Site Safety Program, the following items help reduce the frequency and mitigate worker injuries:

- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Fuel Receipt and Shipping Program (discussed in Section 5.5.3.1)
- Cask/fuel receiving requirements (discussed in Section 5.5.3.2)
- Nuclear Criticality Safety Program (discussed in Section 5.5.3.3)
- Criticality control requirements (discussed in Section 5.5.3.4)

- Basin water level (discussed in Section 5.5.3.12)
- Radiological Protection Program (discussed in Section 5.5.3.14)
- Basins and canal system (discussed in Section 5.5.3.16)
- Storage racks, bundles, buckets, bucket inserts, blocking devices, and associated hardware (discussed in Section 5.5.3.17)

5.5.13 GROUP 4: INJURY TO WORKERS (INDUSTRIAL-RELATED EVENTS)

The following items help reduce the frequency and mitigate worker injuries:

- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Fuel Receipt and Shipping Program (discussed in Section 5.5.3.1)
- Cask/fuel receiving requirements (discussed in Section 5.5.3.2)
- Basin Water Chemistry Control Program (discussed in Section 5.5.3.13)
- Radiological Protection Program (discussed in Section 5.5.3.14)
- Reactor Deionizer Resin Replacement/Regeneration Program (discussed in Section 5.5.6.1)

5.5.14 GROUP 5: CRITICALITY

Sections 5.5.3 and 5.5.5 discuss process- and NPH-induced criticality controls.

5.5.15 GROUP 6: INTERFACES

The following items help reduce the frequency and mitigate interface events:

- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Fuel Receipt and Shipping Program (discussed in Section 5.5.3.1)
- Fire Protection Program (discussed in Section 5.5.10.1)

5.5.16 GROUP 7: WASTE DISPOSAL

The following programs help reduce the frequency and mitigate waste disposal events:

- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Radiological Protection Program (discussed in Section 5.5.3.14)
- Waste Management Program (discussed in Section 5.5.11.1)

5.5.17 GROUP 8: MAINTENANCE

The following programs help reduce the frequency and mitigate maintenance events:

- Event Response Procedures Including Abnormal Operating Procedures, Alarm Response Procedures and the Emergency Preparedness Program (discussed in Section 5.5.1.2)
- Worker Training and Qualification Program (discussed in Section 5.5.1.3)
- Radiological Protection Program (discussed in Section 5.5.3.14)
- Hazardous Material Inventory Control Program (discussed in Section 5.5.1.1)
- Fire Protection Program (discussed in Section 5.5.10.1)

5.5.18 GROUP 9: EXTRINSIC HAZARDS

Section 5.5.2 discusses surface vehicle impact controls.

5.5.19 GROUP 10: NATURAL PHENOMENA

Sections 5.5.4 and 5.5.5 discuss severe weather and earthquake events.

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5.6 DESIGN FEATURES

DOE Order 5480.22 (Ref. 3) defines design features as “those features not covered elsewhere in the TSR that, if altered or modified, would have a significant effect on safety.” The three areas required to be addressed by DOE Order 5480.22 are vital passive components, configuration and physical arrangement, and materials. Chapters 2 and 4 provide details on the design features and their safety function. Design features of specific concern for the TSR are the following:

- Fuel basins and canal system
- Basin water level indicator
- Fuel storage racks and associated hardware
- Research reactor fuel counter safety skirt
- Shielding (Basin walls and control room concrete wall)
- SRS 70-ton casks
- Building 244-H (includes 100-Ton crane south travel mechanical stop)
- Berm at Chemical pad No. 2
- Nitric acid tank
- Sodium nitrite tank
- Sodium nitrite hold tank
- Fuel transfer and storage equipment (i.e., bundles, buckets, bucket inserts, and blocking devices)

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5.7 INTERFACE WITH TSRs FROM OTHER FACILITIES

All facilities that deliver fuel assemblies to the RBOF shall comply with the Fuel Receipt and Shipping Program established by the RBOF TSR. When shipping fuel assemblies to other facilities, the RBOF and RRF shall meet the authorization basis assumptions of the receiving facility. Administrative Controls shall be established to ensure that the contents of waste tank EP38 meet the WAC of the receiving facility. This verification shall be performed prior to transfer to the receiving facility. Administrative Controls, through the Reactor Waste Management Program, shall be established to ensure that RBOF waste being transported by waste trailers is controlled.

The Reactor Deionizer Resin Replacement/Regeneration Program and Reactor Basin Water Chemistry Control Program shall control the activity of the resin and water in the portable deionizers received from other facilities. These programs will be controlled by the applicable facility's TSR.

The transfer lines from the RBOF to the tank farm shall be controlled by the tank farm Configuration Control Program to ensure that changes to the lines do not affect the shielding provided by being underground when waste is transferred.

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5.8 REFERENCES

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9. Functional Classification Report for the Receiving Basin for Offsite Fuels and Resin Regeneration Facility (U). WSRC-TR-96-0093, Rev. 1, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC, September 1997.
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5.9 TABLES

Table 5.3.1-1 Chemical Spills

ACCIDENT: Loss of 50% Sodium Hydroxide
SYSTEM: Event 316

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Hazardous Material Inventory Control Program	SC	All Modes	Administrative Control	Public and Worker Protection - Exceeding the Maximum Allowable Inventory could challenge EG for public and worker protection. Program shall establish maximum inventory limits for hazardous material within the facility (1,100 gallons of sodium hydroxide)	FC 5.2.2 FC 5.4.9
Worker Training and Qualification Program	SS	All Modes	Administrative Control	Worker Protection - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	Worker Protection - To provide proper response (i.e., evacuation) to Emergency Conditions.	FC 5.4.4

STAFFING
REQUIREMENTS: None

Table 5.3.1-1 Chemical Spills (continued)

ACCIDENT: Loss of 50% Nitric Acid

SYSTEM: Event 318

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Hazardous Material Inventory Control Program	SC	All Modes	Administrative Control	<u>Public and Worker Protection</u> - Exceeding the Maximum Allowable Inventory could challenge EG for public and worker protection. Program shall establish maximum inventory limits for hazardous material within the facility (1,100 gallons of nitric acid)	FC 5.2.2 FC 5.4.9
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i.e., evacuation) to Emergency Conditions.	FC 5.4.4

STAFFING
REQUIREMENTS: None

Table 5.3.1-1 Chemical Spills (continued)

ACCIDENT: Loss of 30% Sodium Nitrite
SYSTEM: Event 314a

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Hazardous Material Inventory Control Program	SC	All Modes	Administrative Control	Public and Worker Protection - Exceeding the Maximum Allowable Inventory could challenge EG for public and worker protection. Program shall establish maximum inventory limits for hazardous material within the facility (900 gallons of sodium nitrite)	FC 5.2.2 FC 5.4.9
Chemical pad No. 2 berm, and drain system to waste tank EP38 design	SS	All Modes	Design Feature	Worker Protection - To ensure sodium nitrite and nitric acid do not mix. The mixing of sodium nitrite and nitric acid could cause generation of nitrous dioxide.	FC 5.3.11
Operating procedures for the chemical pad No. 2 pump.	SS	All Modes	Administrative Control	Worker Protection - To prevent pumping the contents of chemical pad No. 2 to waste tank EP38 without first sampling for nitric acid which ensures sodium nitrite and nitric acid do not mix.	FC 5.4.12
Worker Training and Qualification Program	SS	All Modes	Administrative Control	Worker Protection - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	Worker Protection - To provide proper response (i.e., evacuation) to Emergency Conditions.	FC 5.4.4

STAFFING
REQUIREMENTS: None

Table 5.3.1-2 Surface Vehicle Impact

ACCIDENT: Loss of 50% Nitric Acid					
SYSTEM: Event 345c					
Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Hazardous Material Inventory Control Program	SC	All Modes	Administrative Control	<u>Public and Worker Protection</u> - Exceeding the Maximum Allowable Inventory could challenge EG for public and worker protection. Program shall establish maximum inventory limits for hazardous material within the facility (1,100 gallons of nitric acid)	FC 5.2.2 FC 5.4.9
Facility Design - Chemical Tanks	SS	All Modes	Design Feature	<u>Worker Protection</u> - Physical location reduces the likelihood of chemical tanks being struck by vehicles	FC 5.3.10 FC 5.3.12 FC 5.3.13
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions	FC 5.4.4
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8

STAFFING
REQUIREMENTS: None

Table 5.3.1-2 Surface Vehicle Impact (continued)

ACCIDENT: Loss of 30% Sodium Nitrite
SYSTEM: Event 345d

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Hazardous Material Inventory Control Program	SC	All Modes	Administrative Control	Public and Worker Protection - Exceeding the Maximum Allowable Inventory could challenge EG for public and worker protection. Program shall establish maximum inventory limits for hazardous material within the facility (900 gallons of sodium nitrite)	FC 5.2.2 FC 5.4.9
Facility Design - Chemical Tanks	SS	All Modes	Design Feature	Worker Protection - Physical location reduces the likelihood of chemical tanks being struck by vehicles	FC 5.3.10 FC 5.3.12 FC 5.3.13
Worker Training and Qualification Program	SS	All Modes	Administrative Control	Worker Protection - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	Worker Protection - To provide proper response (i.e., evacuation) to Emergency Conditions.	FC 5.4.4

STAFFING
REQUIREMENTS: None

Table 5.3.1-3 Process Induced Criticality

ACCIDENT: Criticality

SYSTEM: Events 24, 34, 35, 36, 37a, 37b, 38, 39, 40, 40b, 40a, 41, 42, 42a, 44, and 404 address several accident initiators but the items of concern are applicable to all events. Specific controls/attributes related to a specific event are noted in the Basis column

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Onsite Cask	SS	All Modes	Design Feature	<u>Worker Protection</u> - Containment design of the cask required.	FC 5.3.5
70-Ton Cask Shipping Requirements	SS	All Modes	LCO	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides. SRS 70-ton cask shipping requirements ensure that cadmium plates and cask slot blocking devices are installed in accordance with the applicable NSDS. Also, the requirements ensure that the cadmium plates have passed a neutron transmission test.	FC 5.2.1 FC 5.4.1
Cask/fuel receiving requirements	SS	All Modes	LCO	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides by ensuring that the fuel identification matches the cask loading diagram (or DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses.	FC 5.2.1 FC 5.4.1
Basins and Canal Systems	SC	All Modes	Design Feature	<u>Public and Worker Protection</u> - To provide primary containment for water.	FC 5.1.1 FC 5.3.2
100-Ton Crane South Travel Limit Switch System	SS	All Modes	LCO	<u>Worker Protection</u> - To prevent the crane from going over #2 storage basin and contacting mechanical stops; thus, preventing the crane load from falling into the fuel storage basin initiating or exacerbating certain criticality events. (Event 37a)	FC 5.3.8

Table 5.3.1-3 Process Induced Criticality (continued)

ACCIDENT: Criticality (continued)

SYSTEM: Events 24, 34, 35, 36, 37a, 37b, 38, 39, 40, 40b, 40a, 41, 42, 42a, 44, and 404 address several accident initiators but the items of concern are applicable to all events. Specific controls/attributes related to a specific event are noted in the Basis column

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Fuel Receipt and Shipping Program	SC	All Modes	Administrative Controls	<p>Public and Worker Protection - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides. Program specifics include:</p> <p>Fuel meets AB assumptions (isotopic characteristics, minimum cooling time, maximum heat load, maximum radionuclide inventory)</p> <p>The onsite cask to be processed with water unless determined to be safe through the USQ process (Events 24 and 42a).</p> <p>The casks have documentation (offsite: Safety Analysis Report for Packaging and Certificate of Compliance, onsite: WSRC Procedure Manual 19Q). The casks are loaded such that the fuel is in a critically safe arrangements</p> <p>Controls to ensure flammable antifreeze is not purged to waste tank EP-38. Controls to address casks deflagration concern. Controls to restrict incoming fuel containing any material that will violently react with water (e.g., sodium or potassium). Fuel and its containers have sufficient negative buoyancy. The number of casks received, fuel elements received, shipments from the RBOF, number of operations, etc., are tracked to meet the assumptions of the frequency analyses of the criticality events. Fuel integrity must be confirmed prior to shipping fuel to the RBOF. Fuel meets storage rack weight requirements of the structural analysis.</p>	FC 5.2.1 FC 5.4.1

Table 5.3.1-3 Process Induced Criticality (continued)

ACCIDENT: Criticality (continued)

SYSTEM: Events 24, 34, 35, 36, 37a, 37b, 38, 39, 40, 40b, 40a, 41, 42, 42a, 44, and 404 address several accident initiators but the items of concern are applicable to all events. Specific controls/attributes related to a specific event are noted in the Basis column

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Hoisting and Rigging Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure that the crane and its associated equipment is maintained which minimizes its likelihood of failure.	FC 5.4.1.3
Crane South End Mechanical Stops	SS	All Modes	Design Feature	<u>Worker Protection</u> - To prevent from traveling beyond the end of the crane rails.	FC 5.3.8
Shielding/ Basin Walls and Control Room Walls	SS	All Modes	Design Feature	<u>Worker Protection</u> - To provide protection to workers located outside RBOF and in the control room	FC 5.3.4
RRFC Skirt	SS	Fuel Handling	Design Feature	<u>Worker Protection</u> - To prevent fuel being placed below the RRFC	FC 5.3.7
Nuclear Criticality Safety Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To prevent fuel canning infractions, rack or fuel snagging, bundling infractions, fuel drops and spills, and unauthorized disassembly operations. Also, the Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. The NSDSs define the allowable storage configurations and storage hardware, as well as the allowable neutronic interaction with other nearby fuels.	FC 5.4.2
Basin Water Chemistry Control Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure that the chemistry of the basin is favorable for long-term storage of fuel and that only minimal corrosion of the fuel occurs. In addition, the program ensures that the alpha activity of the basin water remains within the analyzed amount such that fissile materials will not accumulate to a critical mass in the basin water filtration and deionizer system. (Events 39, 404)	FC 5.4.5

Table 5.3.1-3 Process Induced Criticality (continued)

ACIDENT: Criticality (continued)

SYSTEM: Events 24, 34, 35, 36, 37a, 37b, 38, 39, 40, 40b, 40a, 41, 42, 42a, 44, and 404 address several accident initiators but the items of concern are applicable to all events. Specific controls/attributes related to a specific event are noted in the Basis column

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Storage Racks and Associated Hardware	SS	All Modes	Design Feature	<u>Worker Protection</u> - To maintain the subcritical spacing of the fuel assemblies stored in RBOF during normal operations, and following earthquakes.	FC 5.3.1
Bundles, Buckets, Bucket Insert and Blocking Devices	SS	All Modes	Design Features	<u>Worker Protection</u> - To maintain the subcritical spacing of the fuel assemblies stored in RBOF during normal operations, and following earthquakes.	FC 5.3.15
Deionizer Resin Replacement / Regeneration Program	SS	All Modes	Administrative Controls	<u>Worker Protection</u> - To ensure the RBOF Deionizer Bed resin is periodically monitored, regenerated or replaced consistent with the radioisotopic buildup assumptions of the SAR.	FC 5.4.6
Criticality Control Requirements	SS	All Modes	LCO	<u>Worker Protection</u> - To control fuel handling and storage in accordance with the applicable NSDS. The following also serve to reduce the frequency of this event: fuel storage in isolation, neutron reflectors, support grating, and cask basket unloading with fuel units greater than the applicable safe number (unless authorized by USQ).	FC 5.4.2
NIM System	SS	All Modes	LCO	<u>Worker Protection</u> - To provide a means of alerting the RBOF worker to a criticality with fission product release. The function of the NIM is to alarm thus allowing the workers to reduce exposure to direct radiation, fission product gases, shine or inhalation	FC 5.3.6

Table 5.3.1-3 Process Induced Criticality (continued)

ACCIDENT:	Criticality (continued)
SYSTEM:	Events 24, 34, 35, 36, 37a, 37b, 38, 39, 40, 40b, 40a, 41, 42, 42a, 44, and 404 address several accident initiators but the items of concern are applicable to all events. Specific controls/attributes related to a specific event are noted in the Basis column

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3
Basin Water Level	SC	All Modes	LCO	<u>Public and Worker Protection</u> - To ensure water level is maintained such that the Basin and Associated Canals can provide shielding and a decontamination factor.	FC 5.1.2 FC 5.3.3
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions	FC 5.4.4

STAFFING
REQUIREMENTS: None

Table 5.3.1-4 Severe Weather

ACCIDENT: Chemical Spills (Freezing Temperatures, etc.)

SYSTEM: Events 347, 348

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Hazardous Material Inventory Control Program	SC	All Modes	Administrative Control	<u>Public and Worker Protection</u> - Exceeding the Maximum Allowable Inventory could challenge MOI and worker EG. Program shall establish maximum inventory limits for hazardous material within the facility (1,100 gals HNO ₃ , 1,100 gals NaOH, 900 gals NaNO ₂)	FC 5.2.2 FC 5.4.9
Chemical pad No. 2 berm, and drain system to waste tank EP38 design	SS	All Modes	Design Feature	<u>Worker Protection</u> - To ensure sodium nitrite and nitric acid do not mix. The mixing of sodium nitrite and nitric acid could generate of nitrous dioxide.	FC 5.3.11
Operating procedures for the chemical pad No. 2 pump	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To prevent pumping the contents of chemical pad No. 2 to waste tank EP38 without first sampling for nitric acid which ensures sodium nitrite and nitric acid do not mix.	FC 5.4.12
Sodium Nitrite and Nitric Acid Chemical Tanks	SS	All Modes	Design Feature	<u>Worker Protection</u> - To ensure that the inventories of the chemicals do not mixed by remaining on their respective chemical pad during NPH events.	FC 5.3.10 FC 5.3.12 FC 5.3.13
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions	FC 5.4.4
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8

STAFFING
REQUIREMENTS: None

Table 5.3.1-4 Severe Weather (continued)

ACCIDENT: High Wind Induced Criticality		Events 64, 65			
SYSTEM:					
Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Building 244-H	SS	All Modes	Design Feature	Worker Protection - To withstand the NPH event according to the fragility curve used in the frequency analysis	FC 5.3.9
Storage Racks and Associated Hardware	SS	All Modes	Design Feature	Worker Protection - To maintain the subcritical spacing of the fuel assemblies stored in RBOF.	FC 5.3.1
Bundles, Buckets, Bucket Insert and Blocking Devices	SS	All Modes	Design Features	Worker Protection - To maintain the subcritical spacing of the fuel assemblies stored in RBOF.	FC 5.3.15
Basins and Canal Systems	SC	All Modes	Design Feature	Public and Worker Protection - To provide primary containment for water.	FC 5.1.1 FC 5.3.2
100-Ton Crane South Travel Limit Switch System	SS	All Modes	LCO	Worker Protection - To prevent the crane from going over #2 storage basin and contacting mechanical stops; thus, preventing the crane load from falling into the fuel storage basin initiating or exacerbating certain criticality events.	FC 5.3.8
Hoisting and Rigging Program	SS	All Modes	Administrative Control	Worker Protection - To ensure that the crane and its associated equipment is maintained which minimizes its likelihood of failure.	FC 5.4.13
Shielding/ Basin Concrete Walls and Control Room Walls	SS	All Modes	Design Feature	Worker Protection - To provide protection to workers located outside RBOF and in the control room	FC 5.3.4
Crane South End Mechanical Stops	SS	All Modes	Design Feature	Worker Protection - To prevent the 100-ton crane from traveling beyond the end of the crane rails.	FC 5.3.9

Table 5.3.1-4 Severe Weather (continued)

ACCIDENT: High Wind Induced Criticality (continued)

SYSTEM: Events 64, 65

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Fuel Receipt and Shipping Program	SC	All Modes	Administrative Controls	<p><u>Public and Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides. Program specifics include:</p> <p>Fuel meets AB assumptions (isotopic characteristics, minimum cooling time, maximum heat load, maximum radionuclide inventory)</p> <p>The casks have documentation (offsite: Safety Analysis Report for Packaging and Certificate of Compliance) or design (onsite: WSRC Procedure Manual 19Q). The casks are loaded such that the fuel is in a critically safe arrangements.</p>	FC 5.2.1 FC 5.4.1
Cask/fuel receiving requirements	SS	All Modes	LCO	<p><u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides by ensuring that the fuel identification matches the cask loading diagram (or DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses.</p>	FC 5.2.1
Nuclear Criticality Safety Program	SS	All Modes	Administrative Control	<p><u>Worker Protection</u> - The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. The NSDSs define the allowable storage configurations and storage hardware, as well as the allowable neutronic interaction with other nearby fuels.</p>	FC 5.4.2
Criticality Control Requirements	SS	All Modes	LCO	<p><u>Worker Protection</u> - To control fuel handling and storage in accordance with the applicable NSDS. The following also serve to reduce the frequency of this event: fuel storage in isolation, neutron reflectors, support grating, and cask basket unloading with fuel units greater than the applicable safe number (unless authorized by USQ).</p>	FC 5.4.2

Table 5.3.1-4 Severe Weather (continued)

ACCIDENT: High Wind Induced Criticality (continued)

SYSTEM: Events 64, 65

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3
Basin Water Level	SC	All Modes	LCO	<u>Public and Worker Protection</u> - To ensure water level is maintained such that the Basin and Associated Canals can provide shielding and a decontamination factor.	FC 5.1.2 FC 5.3.3
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions	FC 5.4.4

STAFFING
REQUIREMENTS: None

Table 5.3.1-4 Severe Weather (continued)

ACCIDENT: Fuel Breach (Seismic and High Wind Initiated)

SYSTEM: Event 18

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Building 244-H	SS	All Modes	Design Feature	<u>Worker Protection</u> - To withstand the NPH event according to the fragility curve used in the frequency analysis	FC 5.3.9
Storage Racks and Associated Hardware	SS	All Modes	Design Feature	<u>Worker Protection</u> - To maintain the subcritical spacing of the fuel assemblies stored in RBOF during normal operations, and following earthquakes.	FC 5.3.1
Bundles, Buckets, Bucket Insert and Blocking Devices	SS	All Modes	Design Features	<u>Worker Protection</u> - To maintain the subcritical spacing of the fuel assemblies stored in RBOF during normal operations, and following earthquakes.	FC 5.3.15
Cask/fuel receiving requirements	SS	All Modes	LCO	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides by ensuring that the fuel identification matches the cask loading diagram (or DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses.	FC 5.2.1
100-Ton Crane South Travel Limit Switch System	SS	All Modes	LCO	<u>Worker Protection</u> - To prevent the crane from going over #2 storage basin and contacting mechanical stops; thus, preventing the crane load from falling into the fuel storage basin initiating or exacerbating certain criticality events.	FC 5.3.8
Hoisting and Rigging Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure that the crane and its associated equipment is maintained which minimizes its likelihood of failure.	FC 5.4.13
Crane South End Mechanical Stops	SS	All Modes	Design Feature	<u>Worker Protection</u> - To prevent the 100-ton crane from traveling beyond the end of the crane rails.	FC 5.3.9

Table 5.3.1-4 Severe Weather (continued)

ACCIDENT: Fuel Breach (Seismic and High Wind Initiated) (continued)

SYSTEM: Event 18

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Nuclear Criticality Safety Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. The NSDSs define the allowable storage configurations and storage hardware, as well as the allowable neutronic interaction with other nearby fuels	FC 5.4.2
Criticality Control Requirements	SS	All Modes	LCO	<u>Worker Protection</u> - To control fuel handling and storage in accordance with the applicable NSDS. The following also serve to reduce the frequency of this event: fuel storage in isolation, neutron reflectors, support grating, and cask basket unloading with fuel units greater than the applicable safe number (unless authorized by USQ).	FC 5.4.2
Fuel Receipt and Shipping Program	SC	All Modes	Administrative Controls	<u>Public and Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides. Program specifics include: Fuel meets AB assumptions (isotopic characteristics, minimum cooling time, maximum heat load, maximum radionuclide inventory)	FC 5.2.1 FC 5.4.1
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring and particulate air filter monitoring.	FC 5.4.3

Table 5.3.1.4 Severe Weather (continued)

ACCIDENT: Fuel Breach (Seismic and High Wind Initiated) (continued)

SYSTEM: Event 18

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Basins and Canal Systems	SC	All Modes	Design Feature	<u>Public and Worker Protection</u> - To provide primary containment for water (seismic only).	FC 5.1.1 FC 5.3.2
Basin Water Level	SC	All Modes	LCO	<u>Public and Worker Protection</u> - To ensure water level is maintained such that the Basin and Associated Canals can protect fuel from missiles	FC 5.1.2 FC 5.2.3 FC 5.3.3
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions	FC 5.4.4

STAFFING
REQUIREMENTS: None

Table 5.3.1-5 Earthquake

ACCIDENT: Seismic Induced Criticality

SYSTEM: Event 63

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Building 244-H	SS	All Modes	Design Feature	<u>Worker Protection</u> - To withstand the NPH event according to the fragility curve used in the frequency analysis	FC 5.3.9
Storage Racks and Associated Hardware	SS	All Modes	Design Feature	<u>Worker Protection</u> - To maintain the subcritical spacing of the fuel assemblies stored in RBOF during normal operations, and following earthquakes.	FC 5.3.1
Bundles, Buckets, Bucket Insert and Blocking Devices	SS	All Modes	Design Features	<u>Worker Protection</u> - To maintain the subcritical spacing of the fuel assemblies stored in RBOF during normal operations, and following earthquakes.	FC 5.3.15
Basins and Canal Systems	SC	All Modes	Design Feature	<u>Public and Worker Protection</u> - To provide primary containment for water (seismic only).	FC 5.1.1 FC 5.3.2
100-Ton Crane South Travel Limit Switch System	SS	All Modes	LCO	<u>Worker Protection</u> - To prevent the crane from going over #2 storage basin and contacting mechanical stops; thus, preventing the crane load from falling into the fuel storage basin initiating or exacerbating certain criticality events.	FC 5.3.8
Hoisting and Rigging Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure that the crane and its associated equipment is maintained which minimizes its likelihood of failure.	FC 5.4.13
Crane South End Mechanical Stops	SS	All Modes	Design Feature	<u>Worker Protection</u> - To prevent from traveling beyond the end of the crane rails.	FC 5.3.9
Cask/fuel receiving requirements	SS	All Modes	LCO	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides by ensuring that the fuel identification matches the cask loading diagram (or DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses.	FC 5.2.1

Table 5.3.1-5 Earthquake (continued)

ACCIDENT: Seismic Induced Criticality (continued)
 SYSTEM: Event 63

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Fuel Receipt and Shipping Program	SC	All Modes	Administrative Controls	<p>Public and Worker Protection - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides. Program specifics include:</p> <p>Fuel meets AB assumptions (isotopic characteristics, minimum cooling time, maximum heat load, maximum radionuclide inventory)</p> <p>The casks have documentation (offsite: Safety Analysis Report for Packaging and Certificate of Compliance) or design (onsite: WSRC Procedure Manual 19Q). The casks are loaded such that the fuel is in a critically safe arrangements. Fuel meets storage rack weight requirements of the structural analysis.</p>	FC 5.2.1 FC 5.4.1
Nuclear Criticality Safety Program	SS	All Modes	Administrative Control	<p><u>Worker Protection</u> - The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. The NSDSs define the allowable storage configurations and storage hardware, as well as the allowable neutronic interaction with other nearby fuels.</p>	FC 5.4.2
Criticality Control Requirements	SS	All Modes	LCO	<p><u>Worker Protection</u> - To control fuel handling and storage in accordance with the applicable NSDS. The following also serve to reduce the frequency of this event: fuel storage in isolation, neutron reflectors, support grating, and cask basket unloading with fuel units greater than the applicable safe number (unless authorized by USQ).</p>	FC 5.4.2
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<p><u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs</p>	FC 5.4.8

STAFFING
 REQUIREMENTS: None

Table 5.3.1-5 Earthquake (continued)

ACCIDENT: Breach of Pool Basin
 SYSTEM: Events 43 and 63a

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Basins and Canal Systems	SC	All Modes	Design Feature	<u>Public and Worker Protection</u> - To provide primary containment for radioactive contaminants and maintain water level allowing only minor basin water leakage following an earthquake.	FC 5.1.1 FC 5.3.2
Basin Water Chemistry Control Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure that the chemistry of the basin is favorable for long-term storage of fuel and that only minimal corrosion of the fuel occurs.	FC 5.4.5
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions	FC 5.4.4

STAFFING
 REQUIREMENTS: None

Table 5.3.1-5 Earthquake (continued)

ACCIDENT: Breach of Waste Tank EP-38
SYSTEM: Event 401

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Deionizer Resin Replacement/Regeneration Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To establish programmatic controls to ensure the RBOF Deionizer bed is periodically monitored, replaced or regenerated consistent with radioisotopic buildup assumptions of the Safety Analysis.	FC 5.4.6
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring.	FC 5.4.3
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Design Feature	<u>Worker Protection</u> - To provide proper response (i.e. evacuation) to Emergency Conditions	FC 5.4.4
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8

STAFFING
REQUIREMENTS: None

Table 5.3.1-5 Earthquake (continued)

ACCIDENT: Loss of Chemical Dumpster Containment

SYSTEM: Event 347

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Hazardous Material Inventory Control Program	SC	All Modes	Administrative Control	Public and Worker Protection - Exceeding the Maximum Allowable Inventory could challenge MOI and worker EG. Program shall establish maximum inventory limits for hazardous material within the facility (1,100 gallons HNO ₃ , 1,100 gallons NaOH, 900 gallons NaNO ₂)	FC 5.2.2 FC 5.4.9
Chemical pad No. 2 berm, and drain system to waste tank EP38 design	SS	All Modes	Design Feature	Worker Protection - To ensure sodium nitrite and nitric acid do not mix. The mixing of sodium nitrite and nitric acid could generate of nitrous dioxide.	FC 5.3.11
Operating procedures for the chemical pad No. 2 pump.	SS	All Modes	Administrative Control	Worker Protection - To prevent pumping the contents of chemical pad No. 2 to waste tank EP38 without first sampling for nitric acid which ensures sodium nitrite and nitric acid do not mix.	FC 5.4.12
Worker Training and Qualification Program	SS	All Modes	Administrative Control	Worker Protection - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	Worker Protection - To provide proper response (i.e., evacuation) to Emergency Conditions	FC 5.4.4
Sodium Nitrite and Nitric Acid Chemical Tanks	SS	All Modes	Design Feature	Worker Protection - To ensure that the inventories of the chemicals do not mix by remaining on their respective chemical pad during NPH event.	FC 5.3.10 FC 5.3.12 FC 5.3.13

STAFFING
REQUIREMENTS: None

Table 5.3.1-6 Fire

ACCIDENT: Resin Fire Accident

SYSTEM: Event 310, Portable Deionizer

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Safety</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring.	FC 5.4.3
Reactor Deionizer Resin Replacement/Regeneration Program	SS	All Modes	Administrative Control	<u>Worker Safety</u> - To control portable deionizer radiological activity limits	FC 5.4.6
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i.e., evacuation) to Emergency Conditions	FC 5.4.4
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8

STAFFING
REQUIREMENTS: None

Table 5.3.1-7 Explosion

ACCIDENT: Explosion Involving Nitric Acid and Resin					
SYSTEM: Event 400a, 329c					
Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3
Deionizer Resin Replacement/Regeneration Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To establish programmatic controls to ensure the RBOF Deionizer bed is periodically monitored, replaced or regenerated consistent with radioisotopic buildup assumptions of the Safety Analysis.	FC 5.4.6
Reactor Deionizer Resin Replacement/Regeneration Program	SS	All Modes	Administrative Control	<u>Worker Safety</u> - To control portable deionizer radiological activity limits	N/A
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i.e., evacuation) to Emergency Conditions	FC 5.4.4

STAFFING
REQUIREMENTS: None

Table 5.3.1-8 Uncontrolled Chemical Reaction

ACCIDENT: Uncontrolled Chemical Reaction Sodium Fuel and Water

SYSTEM: Event 51

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Fuel Receipt and Shipping Program	SC	All Modes	Administrative Controls	Public and Worker Protection - For this event this program shall restrict the incoming fuel from containing any material which reacts violently with water (e. g., sodium or potassium).	FC 5.2.1 FC 5.4.1
Cask/fuel receiving requirements	SS	All Modes	LCO	Worker Protection - To ensure that the fuel identification matches the cask loading diagram (or DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses.	FC 5.2.1
Radiological Protection Program	SS	All Modes	Administrative Control	Worker Protection - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	Worker Protection - To provide proper response (i. e., evacuation) to Emergency Conditions	FC 5.4.4
Worker Training and Qualification Program	SS	All Modes	Administrative Control	Worker Protection - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8

STAFFING
REQUIREMENTS: None

Table 5.3.1-9 Aircraft Impact

ACCIDENT: Missile Impact
SYSTEM: * Evaluated and shown to be BEU

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
NA	NA	NA	NA	NA	NA

STAFFING
REQUIREMENTS: None

Table 5.3.2-1 Group 1: Fire/ Explosion

ACCIDENT: Fire/ Explosion

SYSTEM: Events 2a, 3, involving casks

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Onsite Cask	SS	All Modes	Design Feature	Event 2a requires cask containment design to be maintained. This event involves the potential for a fire around cask.	FC 5.3.5
Fuel Receipt and Shipping Program	SC	All Modes	Administrative Controls	<u>Worker Protection</u> - To control required to address potential for flammable gas generation in sealed casks containing failed fuel. The casks have documentation (offsite: Safety Analysis Report for Packaging and Certificate of Compliance) or design (onsite: WSRC Procedure Manual 19Q). The casks are loaded such that the fuel is in a critically safe arrangements	FC 5.2.1 FC 5.4.1
Fire Protection Program	SS	All Modes	Administrative Control	<u>Worker Safety</u> - To ensure that appropriate fire protection measures are maintained to protect the facility from fire and ensures the capability to maintain the facility in a safe condition in an fire event.	FC 5.4.7
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions	FC 5.4.4
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8

STAFFING
REQUIREMENTS: None

Table 5.3.2-1 Group 1: Fire/Explosion (continued)

ACCIDENT: Fire/ Explosion		SYSTEM: Events 313, 314 Nitric acid promotes fire and sodium nitrite explosion			
Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Fire Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure that appropriate fire protection measures are maintained to protect the facility from fire and ensures the capability to maintain the facility in a safe condition in an fire event.	FC 5.4.7
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions	FC 5.4.4
Hazardous Material Inventory Control Program	SC	All Modes	Administrative Control	<u>Public and Worker Protection</u> - Exceeding the Maximum Allowable Inventory could challenge MOI and worker EG. Program shall establish maximum inventory limits for hazardous material within the facility (1,100 gallons HNO ₃ , 1,100 gallons NaOH, 900 gallons NaNO ₂)	FC 5.2.2 FC 5.4.9
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8

STAFFING
REQUIREMENTS: None

Table 5.3.2-1 Group 1: Fire/Explosion (continued)

ACCIDENT: Fire/Explosion

SYSTEM: Event 400 involving organic in waste tank from cask containing antifreeze

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Fuel Receipt and Shipping Program	SS	All Modes	Administrative Controls	<u>Worker Safety</u> - The Fuel Receipt and Shipping Program will restrict the draining of flammable antifreeze to waste tank EP-38	FC 5.2.1 FC 5.4.1
Fire Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure that appropriate fire protection measures are maintained to protect the facility from fire and ensures the capability to maintain the facility in a safe condition in an fire event.	FC 5.4.7
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Hazardous Material Inventory Control Program	SC	All Modes	Administrative Control	<u>Public and Worker Protection</u> - Exceeding the Maximum Allowable Inventory could challenge MOI and worker EG. Program shall establish maximum inventory limits for hazardous material within the facility (1,100 gallons HNO ₃ , 1,100 gallons NaOH, 900 gallons NaNO ₂). No antifreeze stored where it can drain into waste tank EP38	FC 5.2.2 FC 5.4.9
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i.e., evacuation) to Emergency Conditions	FC 5.4.4

STAFFING
REQUIREMENTS: None

Table 5.3.2-1 Group 1: Fire/Explosion (continued)

ACCIDENT: Fire/ Explosion		Event 400b Resin fire after drying out high curie loading causes fire, Event 310 is discussed in Table 5.3.1-6					
SYSTEM:		Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
		Basin Water Chemistry Control Program	SS	All Modes	Administrative Control	<u>Worker Safety</u> - To ensure that the activity of the basin water remains within the analyzed amount such that self-ignition of the resin will not occur in the deionizer system.	FC 5.4.5
		Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
		Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i.e., evacuation) to Emergency Conditions	FC 5.4.4
		Radiological Protection Program Radiation Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3
		Deionizer Resin Replacement/Regeneration Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To establish programmatic controls to ensure the RBOF Deionizer bed is periodically monitored, replaced or regenerated consistent with radioisotopic buildup assumptions of the Safety Analysis.	FC 5.4.6

Table 5.3.2-2 Group 2: Loss of Containment

ACCIDENT: Surface Contamination

SYSTEM: Events 4, 23 (damaged or breach of casks on receipt), 9 (basin overflow)

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR / SAR Section
Fuel Receipt and Shipping Program	SC	All Modes	Administrative Controls	Worker Protection - Casks received into the facility are qualified (10 CFR 71 qualified for offsite casks, authorized per site procedures for onsite casks).	FC 5.2.1 FC 5.4.1
Onsite Cask	SS	All Modes	Administrative Control	Worker Protection - To ensure containment design of cask is maintained. Events 4, 23	FC 5.3.5
Basin Water Chemistry Control Program	SS	All Modes	Administrative Control	Worker Protection - To ensure that the activity of the basin water remains within the analyzed amount. Event 9	FC 5.4.5
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	Worker Protection - To provide proper response (i.e., evacuation) to Emergency Conditions. Event 9	FC 5.4.4
Radiological Protection Program	SS	All Modes	Administrative Control	Worker Protection - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3

STAFFING
REQUIREMENTS: None

Table 5.3.2-2 Group 2: Loss of Containment (continued)

ACCIDENT: Subsurface Releases

SYSTEM: Events 8, 8a, 8b (loss of water and effects on fuel in basin), 321 (breach of portable deionizer)

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Fuel Receipt and Shipping Program	SC	All Modes	Administrative Controls	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides. Program specifics include: Fuel meets AB assumptions (isotopic characteristics, minimum cooling time, maximum heat load, maximum radionuclide inventory). Events 8a, 8b	FC 5.4.1
Cask/fuel receiving requirements	SS	All Modes	LCO	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides by ensuring that the fuel identification matches the cask loading diagram (or DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses.	FC 5.2.1
Nuclear Criticality Safety Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To prevent fuel canning infractions, rack or fuel snagging, bundling infractions, fuel drops and spills, and unauthorized disassembly operations. Also, the Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. The NSDSs define the allowable storage configurations and storage hardware, as well as the allowable neutronic interaction with other nearby fuels.	FC 5.4.2
Criticality Control Requirements	SS	All Modes	LCO	<u>Worker Protection</u> - To control fuel handling and storage in accordance with the applicable NSDS. The following also serve to reduce the frequency of this event: fuel storage in isolation, neutron reflectors, support grating, and cask basket unloading with fuel units greater than the applicable safe number (unless authorized by USQ).	FC 5.4.2

Table 5.3.2-2 Group 2: Loss of Containment (continued)

ACCIDENT: Subsurface Releases (continued)

SYSTEM: Events 8, 8a, 8b (loss of water and effects on fuel in basin), 321 (breach of portable deionizer)

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
NIM System	SS	All Modes	LCO	Worker Protection - The function of the NIM is to alarm thus allowing the workers to reduce exposure to direct radiation, fission product gases, shine or inhalation	FC 5.3.6
Remote NIM Alarm	SS	Shutdown	LCO	Worker Protection - To provide a means of alerting H-Area personnel to a potential radiation exposure in RBOF during shutdown	FC 5.3.14
NIM Trouble Portion of Remote 244-H General Alarm	SS	Shutdown	LCO	Worker Protection - To provide a means of alerting H-Area personnel to a problem with the NIMs during RBOF shutdown	FC 5.3.14
Basins and Canal Systems	SC	All Modes	Design Feature	Public and Worker Protection - To provide primary containment for radioactive contaminants and maintain water level allowing only minor basin water leakage following an earthquake. Basins provide structural support for the storage racks and associated hardware.	FC 5.1.1 FC 5.3.2
Storage Racks and Associated Hardware	SS	All Modes	Design Feature	Worker Protection - To maintain the fuel spacing assumed in the criticality and fuel melt analyses during normal operations, and following earthquakes.	FC 5.3.1
Bundles, Buckets, Bucket Insert and Blocking Devices	SS	All Modes	Design Features	Worker Protection - To maintain the subcritical spacing of the fuel assemblies stored in RBOF during normal operations, and following earthquakes.	FC 5.3.15
Radiological Protection Program	SS	All Modes	Administrative Control	Worker Protection - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3

Table 5.3.2-2 Group 2: Loss of Containment (continued)

Subsurface Releases (continued)					
Events 8, 8a, 8b (loss of water and effects on fuel in basin), 32I (breach of portable deionizer)					
Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Hoisting and Rigging Program	SS	All Modes	Administrative Control	Worker Protection - To ensure that the crane and its associated equipment is maintained which minimizes its likelihood of failure.	FC 5.4.13
Reactor Deionizer Resin Replacement/Regeneration Program	SS	All Modes	Administrative Control	Worker Safety - To control portable deionizer radiological activity limits	N/A
Worker Training and Qualification Program	SS	All Modes	Administrative Control	Worker Protection - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Basin Design - Filter/Deionizer System Piping Penetration Configuration	SS	All Modes	Administrative Control	Worker Protection - To ensure basin does not completely drain from F/D system piping leakage.	FC 5.1.1 FC 5.3.2
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	Worker Protection - To provide proper response (i.e., evacuation) to Emergency Conditions	FC 5.4.4
Basin Water Chemistry Control Program	SS	All Modes	Administrative Control	Worker Protection - To limit the basin water activity to 0.1 micro-Curies/ml. Event 8	FC 5.4.5
Reactor Basin Water Chemistry Control Program	SS	All Modes	Administrative Control	Worker Safety - To control portable deionizer radiological activity limits	FC 5.4.15

STAFFING
REQUIREMENTS: None

Table 5.3.2-2 Group 2: Loss of Containment (continued)

ACCIDENT: Subsurface Releases					
SYSTEM: Event 406					
Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Transfer Line Configuration	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure the transfer lines are located underground.	N/A
Radiological Protection Program	SS	All Modes	Administrative	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring.	FC 5.4.3

STAFFING
REQUIREMENTS: None

Table 5.3.2-2 Group 2: Loss of Containment (continued)

ACCIDENT: Airborne Release

SYSTEM: Events 324, 345a (breach of portable deionizer)

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Reactor Basin Water Chemistry Control Program	SS	All Modes	Administrative Control	<u>Worker Safety</u> - To control basin water activity in the portable deionizer	N/A
Reactor Deionizer Resin Replacement/Regeneration Program	SS	All Modes	Administrative Control	<u>Worker Safety</u> - To control portable deionizer radiological activity limits	N/A
Facility Design - Portable Deionizer Location	SS	All Modes	Design Feature	<u>Worker Protection</u> - To reduce the likelihood of the portable deionizer being struck by vehicles	N/A
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i.e., evacuation) to Emergency Conditions	FC 5.4.4
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3

STAFFING
REQUIREMENTS: None

Table 5.3.2-2 Group 2: Loss of Containment (continued)

ACCIDENT: Airborne Release

SYSTEM: Events 5, 5a, 6, 12, 13 (cask handling operation)

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Fuel Receipt and Shipping Program	SC	All Modes	Administrative Controls	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides. Program specifics include: Fuel meets AB assumptions (isotopic characteristics, minimum cooling time, maximum heat load, maximum radionuclide inventory) Casks received into the facility are qualified (10 CFR 71 qualified for offsite casks, authorized per site procedures for onsite casks).	FC 5.2.1 FC 5.4.1
Onsite Cask	SS	All Modes	Design Feature	<u>Worker Protection</u> - To ensure containment design of cask is maintained	FC 5.3.5
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i.e., evacuation) to Emergency Conditions	FC 5.4.4
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3

STAFFING
REQUIREMENTS: None

Table 5.3.2-2 Group 2: Loss of Containment (continued)

ACCIDENT: Airborne Release					
SYSTEM: Events 7, 7a, 10, 10a, 11, 11a (fuel handling)					
Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Fuel Receipt and Shipping Program	SC	All Modes	Administrative Control	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides. Program specifics include: Fuel meets AB assumptions (isotopic characteristics, minimum cooling time, maximum heat load, maximum radionuclide inventory)	FC 5.4.1
Cask/fuel receiving requirements	SS	All Modes	LCO	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides by ensuring that the fuel identification matches the cask loading diagram (or DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses.	FC 5.2.1
Nuclear Criticality Safety Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure the NSDS compliance. <u>Fuel Cropping Operations Controlled</u>	FC 5.4.2
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i.e., evacuation) to Emergency Conditions	FC 5.4.4

Table 5.3.2-2 Group 2: Loss of Containment (continued)

ACCIDENT: Airborne Release (continued)
 SYSTEM: Events 7, 7a, 10, 10a, 11, 11a (fuel handling)

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3

STAFFING
 REQUIREMENTS: None

Table 5.3.2-2 Group 2: Loss of Containment (continued)

ACCIDENT: Airborne Release

SYSTEM: Event 8c

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Fuel Receipt and Shipping Program	SC	All Modes	Administrative Control	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides. Program specifics include: Fuel meets AB assumptions (isotopic characteristics, minimum cooling time, maximum heat load, maximum radionuclide inventory)	FC 5.4.1
Cask/fuel receiving requirements	SS	All Modes	LCO	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides by ensuring that the fuel identification matches the cask loading diagram (or DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses.	FC 5.2.1
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i.e., evacuation) to Emergency Conditions	FC 5.4.4

Table 5.3.2-2 Group 2: Loss of Containment (continued)

ACCIDENT: Airborne Release (continued)
SYSTEM: Event 8c

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
NIM System	SS	All Modes	LCO	Worker Protection - To provide a means of alerting the RBOF worker to a criticality with fission product release. The function of the NIM is to alarm thus allowing the workers to reduce exposure to direct radiation, fission product gases, shine or inhalation	FC 5.3.6
Remote NIM Alarm	SS	Shutdown	LCO	Worker Protection - To provide a means of alerting H-Area personnel to a potential radiation exposure in RBOF during shutdown	FC 5.3.14
NIM Trouble Portion of Remote 244-H General Alarm	SS	Shutdown	LCO	Worker Protection - To provide a means of alerting H-Area personnel to a problem with the NIMs during RBOF shutdown	FC 5.3.14
Hoisting and Rigging Program	SS	All Modes	Administrative Control	Worker Protection - To ensure that the crane and its associated equipment is maintained which minimizes its likelihood of failure.	FC 5.4.13
Basin Water Chemistry Control Program	SS	All Modes	Administrative Control	Worker Protection - To limit the basin water activity to 0.1 micro-Curies/ml. Event 8	FC 5.4.5
Radiological Protection Program	SS	All Modes	Administrative Control	Worker Protection - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3

STAFFING
REQUIREMENTS: None

Table 5.3.2-2 Group 2: Loss of Containment (continued)

ACCIDENT: Airborne Release		Event 402 (waste trailer breach), Event 407 (transfer error to Building 211-H) (Event 401a is covered in Table 5.3.2-10; chemical leaks)			
SYSTEM:					
Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i.e., evacuation) to Emergency Conditions	FC 5.4.4
Waste Management Program	SS	All Modes	Administrative	<u>Worker Protection</u> - To programmatically establish controls to ensure the maximum activity associated with the waste leaving RBOF is maintained.	FC 5.4.11

STAFFING
REQUIREMENTS: None

Table 5.3.2-3 Group 3: Loss of Confinement

ACCIDENT: Loss of Confinement (Ground releases / Filter breach)

SYSTEM: Events 16, 17, 18, 19, 20, 21 (Event 329c bounded by 400a not addressed in this table, discussed in Table 5.3.1-7)

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Fuel Receipt and Shipping Program	SC	All Modes	Administrative Controls	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides. Program specifics include: Fuel meets AB assumptions (isotopic characteristics, minimum cooling time, maximum heat load, maximum radionuclide inventory)	FC 5.2.1
Cask/fuel receiving requirements	SS	All Modes	LCO	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides by ensuring that the fuel identification matches the cask loading diagram (or DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses.	FC 5.4.2
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Nuclear Criticality Safety Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - The Nuclear Criticality Safety Program requires that NSDSs be prepared for one or more specific reactor fuels. The NSDSs define the allowable storage configurations and storage hardware, as well as the allowable neutronic interaction with other nearby fuels.	FC 5.4.2

Table 5.3.2-3 Group 3: Loss of Confinement (continued)

ACCIDENT: Loss of Confinement (Ground releases / Filter breach) (continued)

SYSTEM: Events 16, 17, 18, 19, 20, 21 (Event 329c bounded by 400a not addressed in this table, discussed in Table 5.3.1-7)

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Criticality Control Requirements	SS	All Modes	LCO	<u>Worker Protection</u> - To control fuel handling and storage in accordance with the applicable NSDS. The following also serve to reduce the frequency of this event: fuel storage in isolation, neutron reflectors, support grating, and cask basket unloading with fuel units greater than the applicable safe number (unless authorized by USQ).	FC 5.4.2
Basin Water Level	SC	All Modes	LCO	<u>Public and Worker Protection</u> - To ensure water level is maintained. Provide protection of fuel from missiles	FC 5.1.2 FC 5.2.3 FC 5.3.3
Storage Racks and Associated Hardware	SS	All Modes	Design Feature	<u>Worker Protection</u> - To maintain the subcritical spacing of the fuel assemblies stored in RBOF during normal operations, and following earthquakes.	FC 5.3.1
Bundles, Buckets, Bucket Insert and Blocking Devices	SS	All Modes	Design Features	<u>Worker Protection</u> - To maintain the subcritical spacing of the fuel assemblies stored in RBOF during normal operations, and following earthquakes.	FC 5.3.15
Safety Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide a spotter for trailers backing up into carport.	FC 5.4.10
Basins and Canal Systems	SC	All Modes	Design Feature	<u>Public and Worker Protection</u> - To provide primary containment for radioactive contaminants and maintain water level allowing only minor basin water leakage following an earthquake. Basins provide structural support for the storage racks and associated hardware.	FC 5.1.1 FC 5.3.2

Table 5.3.2-3 Group 3: Loss of Confinement (continued)

ACCIDENT: Loss of Confinement (Ground releases / Filter breach) (continued)

SYSTEM: Events 16, 17, 18, 19, 20, 21 (Event 329c bounded by 400a not addressed in this table, discussed in Table 5.3.1-7)

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Radiological Protection Program	SS	All Modes	Administrative Control	Worker Protection - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	Worker Protection - To provide proper response (i. e., evacuation) to Emergency Conditions	FC 5.4.4

STAFFING
REQUIREMENTS: None

Table 5.3.2-4 Group 4: Injury to Workers

ACCIDENT: Damage To Cask Or Cask Received Outside Shipping Manifest

SYSTEM: Event 25

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Cask/fuel receiving requirements	SS	All Modes	LCO	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides by ensuring that the fuel identification matches the cask loading diagram (or DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses.	FC 5.2.1 FC 5.4.1
Fuel Receipt and Shipping Program	SC	All Modes	Administrative Controls	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides. Program specifics include: Fuel meets AB assumptions (isotopic characteristics, minimum cooling time, maximum heat load, maximum radionuclide inventory)	FC 5.2.1 FC 5.4.1
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions. Provide proper response (i. e., evacuation) to Emergency Conditions	FC 5.4.4

Table 5.3.2-4 Group 4: Injury to Workers (continued)

ACCIDENT: Damage To Cask Or Cask Received Outside Shipping Manifest (continued)
SYSTEM: Event 25

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Worker Training and Qualification Program	SS	All Modes	Administrative Control	Worker Protection - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8

STAFFING
REQUIREMENTS: None

Table 5.3.2-4 Group 4: Injury to Workers (continued)

ACCIDENT: Movement Of Fuel Causing Worker Exposure

SYSTEM: Events 14, 22

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
NIM System	SS	All Modes	LCO	<u>Worker Protection</u> - To provide a means of alerting the RBOF worker to a criticality with fission product release. The function of the NIM is to alarm thus allowing the workers to reduce exposure to direct radiation, fission product gases, shine or inhalation	FC 5.3.6
Fuel Receipt and Shipping Program	SC	All Modes	Administrative Controls	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides. Program specifics include: Fuel meets AB assumptions (isotopic characteristics, minimum cooling time, maximum heat load, maximum radionuclide inventory)	FC 5.2.1 FC 5.4.1
Cask/fuel receiving requirements	SS	All Modes	LCO	<u>Worker Protection</u> - To programmatically establish controls to prevent criticality hazards, and the release of radionuclides by ensuring that the fuel identification matches the cask loading diagram (or DOE Appendix A) so that the characteristics of the fuel received meet the assumptions in the analyses.	FC 5.2.1 FC 5.4.1
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3
Applicable event response procedures.	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions.	FC 5.4.4

STAFFING
REQUIREMENTS: None

Table 5.3.2-4 Group 4: Injury to Workers (continued)

ACCIDENT:	Worker Falls In Basin
SYSTEM:	Event 32

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Basin Water Chemistry Control Program	SS	All Modes	Administrative Control	Worker Protection - To ensure limits on basin water activity is controlled.	FC 5.4.5

STAFFING
REQUIREMENTS: None

Table 5.3.2-4 Group 4: Injury to Workers (continued)

ACCIDENT: Breach of Portable Deionizer					
SYSTEM: Event 334					
Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Reactor Deionizer Resin Replacement/Regeneration Program	SS	All Modes	Administrative Control	<u>Worker Safety</u> - To control portable deionizer radiological activity limits	N/A
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions.	FC 5.4.4

STAFFING
REQUIREMENTS: None

Table 5.3.2-5 Group 5: Criticality

ACCIDENT:	Criticality
SYSTEM:	* Events. 24, 34, 35, 36, 37a, 37b, 38, 39, 40, 40a, 41, 42, 42a, 43, 44, 63, 63a, 64, 65, 404 *

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR / SAR Section
* See Tables 5.3.1-3, 5.3.1-4 and 5.3.1-5 for Process Induced and NPH Criticality	NA	NA	NA	NA	NA

STAFFING
REQUIREMENTS: None

Table 5.3.2-6 Group 6: Interfaces

ACCIDENT: Interface		Events 53, 56, 340 (Event 51 see Table 5.3.1-8)			
Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions.	FC 5.4.4
Fire Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure that appropriate fire protection measures are maintained to protect the facility from fire and ensures the capability to maintain the facility in a safe condition in an fire event.	FC 5.4.7

STAFFING
REQUIREMENTS: None

Table 5.3.2-7 Group 7: Waste Disposal

ACCIDENT: Improper Disposal of Waste
SYSTEM: Events 57, 58, 341, 342

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Waste Management Program	SS	All Modes	Administrative	<u>Worker Protection</u> - To programmatically establish controls to ensure the maximum activity of waste leaving RBOF is maintained.	FC 5.4.11
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions.	FC 5.4.4

STAFFING
REQUIREMENTS: None

Table 5.3.2-8 Group 8: Maintenance

ACCIDENT: Exposure/Contamination Outside Of ALARA

SYSTEM: Events 59, 59a, 60, 343

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions.	FC 5.4.4
Fire Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure that appropriate fire protection measures are maintained to protect the facility from fire and ensures the capability to maintain the facility in a safe condition in an fire event.	FC 5.4.7
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Hazardous Material Inventory Control Program	SC	All Modes	Administrative Control	<u>Public and Worker Protection</u> - Exceeding the Maximum Allowable Inventory could challenge MOI and worker EG. Program shall establish maximum inventory limits for hazardous material within the facility (1,100 gallons HNO ₃ , 1,100 gallons NaOH, 900 gallons NaNO ₂). No flammable antifreeze stored where it can drain into waste tank EP38	FC 5.2.2 FC 5.4.9
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3

STAFFING
REQUIREMENTS: None

Table 5.3.2-9 Group 9: Extrinsic Hazards

ACCIDENT: Vehicle Crashes Chemical Pad No. 1; Hose Box No. 1

SYSTEM: * Events 345c, 345d (See Table 5.3.1-2 Chemical Spills / 345a See Table 5.3.2-2)

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
*	NA	NA	NA	NA	NA

STAFFING
REQUIREMENTS: None

Table 5.3.2-10 Group 10: Natural Phenomena

ACCIDENT: Criticality and Associated Hazards (initiated by Earthquake or Tornado)
SYSTEM: Events 43, 63, 63a, 64, 65*

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
* See Table 5.3.1-4,5.3.1-5	NA	NA	NA	NA	NA

STAFFING
REQUIREMENTS: None

Table 5.3.2.10 Group 10: Natural Phenomena (continued)

ACCIDENT: Chemical Leak (Initiated By Earthquake Tornado/High Wind Rain Shelter Collapse)
 SYSTEM: Events 347, 347a (401a) (Event 348 discussed in Table 5.3.1-4)

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Sodium Nitrite and Nitric Acid Chemical Tanks	SS	All Modes	Design Feature	<u>Worker Protection</u> - To ensure that the inventories of the chemicals do not mixed by remaining on their respective chemical pad during NPH events.	FC 5.3.10 FC 5.3.12 FC 5.3.13
Chemical pad No. 2 berm, and drain system to waste tank EP38 design	SS	All Modes	Design Feature	<u>Worker Protection</u> - To ensure sodium nitrite and nitric acid do not mix. The mixing of sodium nitrite and nitric acid could generation of nitrous dioxide.	FC 5.3.11
Operating procedures for the chemical pad No. 2 pump	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To prevent pumping the contents of chemical pad No. 2 to waste tank EP38 without first sampling for nitric acid which ensures sodium nitrite and nitric acid do not mix. The mixing of sodium nitrite and nitric acid could generation of nitrous dioxide.	FC 5.4.12
Hazardous Material Inventory Control Program	SC	All Modes	Administrative Control	<u>Public and Worker Protection</u> - Exceeding the Maximum Allowable Inventory could challenge MOI and worker EG. Program shall establish maximum inventory limits for hazardous material within the facility. Limit sodium nitrite -900 gal, nitric acid 1,100 gal	FC 5.2.2 FC 5.4.9
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8

Table 5.3.2.10 Group 10: Natural Phenomena (continued)

ACCIDENT: Chemical Leak (Initiated By Earthquake Tornado/High Wind Rain Shelter Collapse) (continued)

SYSTEM: Events 347, 347a (401a) (Event 348 discussed in Table 5.3.1-4)

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Radiological Protection Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide indication to the workers of a high radiation level to permit evacuation by radiation monitoring. Included in this program is providing post event monitoring, particulate air filter monitoring, and surveying casks when received.	FC 5.4.3
Reactor Deionizer Resin Replacement/Regeneration Program	SS	All Modes	Administrative Control	<u>Worker Safety</u> - To control portable deionizer radiological activity limits	N/A
RBOF Deionizer Resin Replacement/Regeneration Program	SS	All Modes	Administrative Control	<u>Worker Safety</u> - To control deionizer radiological activity limits	FC 5.4.6
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.)	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions	FC 5.4.4

STAFFING
REQUIREMENTS: None

Table 5.3.2.10 Group 10: Natural Phenomena (continued)

ACCIDENT: Basin Overflow Following Loss Of Containment Initiated By Floods And/Or Heavy Rains

SYSTEM: Event 66, 67, 69, 349

Item(s) of Concern	Functional Classification	Operating Mode	Operational Limits Required	Basis	FCR Section
Building 244-H	SS	All Modes	Design Feature	<u>Worker Protection</u> - To withstand the NPH event according to the fragility curve used in the frequency analysis	FC 5.3.9
Worker Training and Qualification Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To ensure workers are properly trained to perform duties. This program also provides support to other Safety Significant programs	FC 5.4.8
Basin Water Chemistry Control Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To limit the basin water activity to 0.1 micro-Curies/ml.	FC 5.4.5
Applicable event response procedure (e.g., Emergency Preparedness Program, abnormal operating procedure, or alarm response procedure, etc.) Emergency Response Program	SS	All Modes	Administrative Control	<u>Worker Protection</u> - To provide proper response (i. e., evacuation) to Emergency Conditions. Provide proper response (i. e., evacuation) to Emergency Conditions	FC 5.4.4

STAFFING
REQUIREMENTS: None

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 6

PREVENTION OF INADVERTENT CRITICALITY

AUGUST 1997

UNCLASSIFIED
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NUCLEAR INFORMATION

ADC &
Reviewing
Official:

J.N. Dewes
J.N. DEWES, Manager, Regulatory Programs

Date:

9/23/97

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

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SAVANNAH RIVER SITE

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ACRONYMS

AID	Administration and Infrastructure Division
ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
CMF	Common Mode Failure
CND	Criticality Neutron Dosimeter
CRT	Criticality and Radiation Transport
CSC	Criticality Safety Control
CSEG	Criticality Safety Engineering Group
CSL	Criticality Safety Limit
CTG	Criticality Technology Group
DOE	Department of Energy
DOE-SR	Department of Energy Savannah River Operations Office
DOT	Department of Transportation
EBS	Expanded Basin Storage
ESH&QA	Environmental, Safety, Health, and Quality Assurance
FOSC	Facility Operations Safety Committee
GBA	General Building Area
HFIR	High Flux Isotope Reactor
IAEA	International Atomic Energy Agency
LCO	Limiting Condition for Operation
MP	Management Policy
MRP	Management Requirement and Procedure
MTR	Material Test Reactor
NCSE	Nuclear Criticality Safety Evaluation
NCSRC	Nuclear Criticality Safety Review Committee
NIM	Nuclear Incident Monitor
NOP	Normal Operating Procedure

ACRONYMS (continued)

NPH	Natural Phenomena Hazard
NRC	Nuclear Regulatory Commission
NSC	Nuclear Safety Control
NSDS	Nuclear Safety Data Sheet
PECD	Projects, Engineering, and Construction Division
QAP	Quality Assurance Procedure
RBOF	Receiving Basin for Offsite Fuels
RC&HP	Radiological Control and Health Physics
RCO	Radiological Control Operations
RDP	Reactor Division Procedure
RRF	Resin Regeneration Facility
RRFC	Research Reactor Fuel Counter
SAR	Safety Analysis Report
SFSD	Spent Fuel Storage Division
SRS	Savannah River Site
TSR	Technical Safety Requirement
USQ	Unreviewed Safety Question
WSRC	Westinghouse Savannah River Company

6.0 PREVENTION OF INADVERTENT CRITICALITY

6.1 INTRODUCTION

Management Policy (MP) 4.5 of WSRC-1-01 (Ref. 1) establishes the Nuclear Criticality Safety Program for the Savannah River Site (SRS), which satisfies the requirements of Department of Energy (DOE) Order 5480.24 (Ref. 2). Features of the program are described in WSRC-SCD-3 (Ref. 3) and Chapter 6 of the Generic Safety Analysis Report (SAR) (Ref. 4). DOE has issued DOE Order 420.1 (Ref. 5) as a replacement for DOE Order 5480.24, and Westinghouse Savannah River Company (WSRC) is developing an implementation plan for DOE Order 420.1. Due to the similarity of requirements between the two Orders, the compliance status of the Receiving Basin for Offsite Fuels (RBOF) is expected to be comparable. Since the RBOF contains substantial quantities of fissionable material, the SRS Nuclear Criticality Safety Program is implemented for the RBOF.

WSRC-TR-95-0302 (Ref. 6) establishes the Nuclear Criticality Safety Program and the Fuel Receipt and Shipping Program for the RBOF. The RBOF Nuclear Criticality Safety Program is in compliance with the SRS Nuclear Criticality Safety Program.

6.1.1 APPLICABILITY

This chapter describes the features of the RBOF Nuclear Criticality Safety Program. WSRC-TR-95-0302 (Ref. 6) establishes the Nuclear Criticality Safety Program and the Fuel Receipt and Shipping Program for the RBOF. The presentation of the information in this chapter satisfies the requirements of paragraph 8.b(3)(h) of DOE Order 5480.23 (Ref. 7), as amplified in Attachment 1, paragraph 4.f(3)(d)8, of the Order (Topic 8). The presentation of this information also partially satisfies requirements of DOE Order 5480.23, paragraphs 8.b(3)(b), (f), and (u). However, this chapter is not to be used as the vehicle for review and approval of the Nuclear Criticality Safety Program for the RBOF.

6.1.2 OBJECTIVE

This chapter describes the Nuclear Criticality Safety Program as implemented for the RBOF, which is in compliance with the SRS requirements as provided in MP 4.5 of WSRC-1-01 (Ref. 1), WSRC-SCD-3 (Ref. 3), and Chapter 6 of the Generic SAR (Ref. 4). The policies and requirements of the SRS Nuclear Criticality Safety Program are described in Chapter 6 of the Generic SAR. Only those elements of the program applicable to the RBOF are addressed here. As suggested by DOE-STD-3009-94 (Ref. 8), the RBOF Nuclear Criticality Safety Program is summarized here, including the following:

- Summary descriptions of the RBOF criticality safety design and administrative controls

- General basis and analytical approach used for deriving operational nuclear criticality limits
- Elements and requirements of the Nuclear Criticality Safety Program
- Implementation of criticality control at the RBOF

6.1.3 SCOPE

As indicated in Chapter 6 of the Generic SAR (Ref. 4), all facilities classified as Hazard Category 2 or higher can contain sufficient fissionable materials to present an inadvertent criticality hazard. The RBOF is classified as Hazard Category 2 in accordance with the criteria of DOE-STD-1027-92 (Ref. 9). The RBOF contains fissionable material quantities more than sufficient to present an inadvertent criticality hazard, as shown in Section 3.3.2.2. Thus, a description of the RBOF Nuclear Criticality Safety Program is required in this SAR by DOE Order 5480.23 (Ref. 7). However, this chapter is applicable only to the receipt, handling, and storage basins in Building 244-H and is not applicable to the Resin Regeneration Facility (RRF) and related facilities in Building 244-H (which do not contain fissionable materials other than trace amounts in the resin and resin wastes).

6.1.4 GLOSSARY

The following is a list of terms related to criticality safety in the RBOF; additional terms are available in EFR-RSE-95-0100 (Ref. 10):

BASKET	The (removable) structures within a shipping cask that provide critically safe arrangements of fuel elements or assemblies within a cask, using combinations of relative spacing, positioning, and/or neutron absorbers. Generally referred to as a "cask basket."
BUCKET	A container used to transport individual fuel elements or assemblies or groups (safe fuel unit) of fuel elements or assemblies throughout the basin.
BUNDLE	A container or holder in which fuel is handled, stored, or shipped.
CASK	A general term used to refer to any make or model container authorized for the shipment of spent nuclear fuel.
CASK HANDLING	Cask handling shall be the handling or movement of sealed (i.e., lid secured in place) casks containing fuel in the facility. Venting is permissible.
FACILITY	The areas defined by the combination of RBOF, RRF, and General Building Area (GBA).

FISSILE MATERIAL	Nuclides, or mixtures of nuclides, capable of sustaining a fission chain reaction with thermalized (slow) as well as high energy (fast) neutrons (e.g., U-233, U-235, Pu-239, Pu-241, Am-242, Cm-243, and Cm-245).
FUEL	For the RBOF basin, fuel consists of all SRS fuel and all research reactor fuel.
FUEL HANDLING	The handling or movement of fuel or unsealed casks containing fuel (e.g., unsealing or sealing a cask; unloading or loading a cask; and transferring, inspecting, disassembling, cropping, bundling, and repackaging fuel) in the RBOF basin. Fuel handling shall also include the movement of any submerged device in the RBOF basin within 2 feet of fuel.
FUEL STORAGE	The storage of out-of-cask fuel in its normal storage location, excluding isolated storage. Normal storage is defined in a Nuclear Safety Data Sheet (NSDS).
FUEL STORAGE RACK	Framework for supporting stored fuel under water.
GBA	General Building Area. The areas outside Buildings 244-H and 245-H, including chemical pads No. 1 and No. 2, storage buildings, office buildings, the yard storage areas, and the surrounding grounds.
GRATING	These establish the floor of the tier. They are installed either on the storage rack beams or on top of lugs. The stamped identification designates the width of the trough guides or width of the row and position of the tier.
GUIDE PLATES	These are bolted or welded to the lugs and are provided with swing-down spacers and latching slots to position fuel along the tier. The stamped identification designates row width, tier location, and position in the tier.
ISOLATE/ISOLATED	A fuel piece, a safe fuel unit, or a safe fuel group shall be considered isolated if it is restrained and there is at least 12 inches of water, 60 cm of concrete, or 2 meters of air between it and any other fuel in any direction. Fuel stored in a cask shall be considered isolated.
LUGS	Slots in this component fit over the beams in the storage rack. Other components are attached to the lugs by either bolts or welds. The lug design determines the vertical position of the hardware.
MAJOR REVISION OF NSDS	Any change that is technical in nature or alters requirements.
MINOR REVISION OF NSDS	Typographical error corrections, reference corrections, addition of a note or reference, correction of grammatical errors, and other nontechnical changes that do not alter requirements.

RACK HARDWARE	The combination of guide plates, spacer plates, and gratings designed to provide specific lateral, longitudinal, and vertical spacing between fuel pieces. Hardware is also provided for storage of bundles or tubes containing several fuel pieces. Specific rack hardware is described in EFR-RSE-95-0100 (Ref. 10).
RBOF	Receiving Basin for Offsite Fuels facility (Building 244-H).
REMOVABLE RACK TIE	Metal latches used in place of row rack ties, which are not latched or otherwise do not properly engage.
ROW LOCKS	Device installed at the end of the adjustable hardware rows to prevent movement of fuel into or out of the row. This device can only be removed with a special tool kept under control of supervision.
ROW RACK TIE	Metal latches that maintain horizontal spacing between adjacent I-beams.
ROW WIDTH WITH HARDWARE	The horizontal distance between guide plates within which the fuel pieces are supported in near-vertical position.
ROW WIDTH WITHOUT HARDWARE	The center-to-center horizontal distance between rack I-beams.
RRF	Resin Regeneration Facility (Building 245-H).
SAFE FUEL GROUP	A specified number of safe fuel units stored together in a straight row, one fuel assembly wide, with a calculated neutron multiplication factor (k_{eff}) less than K_{safe} . (K_{safe} is defined to be a conservative upper bound on reactivity, for which the safe fuel group is safely subcritical.)
SAFE FUEL UNIT	A specified number of fuel pieces (e.g., plates, assemblies, rods, oxide canisters, buckets, bundles) stored together in the most reactive configuration with a calculated neutron multiplication factor (k_{eff}) less than K_{safe} . (K_{safe} is defined to be a conservative upper bound on reactivity, for which the safe fuel unit is safely subcritical.)
SPACER PLATES	Latches that engage both north and south guide plates and position fuel along the row in the east west direction.
SPACING IN ROW	The average center-to-center distance between fuel pieces in the same row, determined by the interval between spacer plates (also referred to as along-row spacing).
TEST TUBE STORAGE	Test tube storage consist of small test tube storage and big test tube storage. Small test tube storage provides storage for fuel test tubes. Big test tube storage is used to store fuel suspected of cladding failure.
TIER	One of several rows of fuel, one above another, with Tier 1 being the lowest.

TROUGH GUIDES	Guide plates bolted or welded to the fuel storage racks and used to position the "V" trough at the bottom of a tier.
"V" TROUGH	These are V-shaped troughs that fit between the trough guides and are intended to center the bottom of the fuel in the tier. The stamped identification designates the type of trough, width of trough, and position in the tier. Troughs are stamped for location similar to grating.

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6.2 REQUIREMENTS

The RBOF Nuclear Criticality Safety Program (established in WSRC-TR-95-0302 [Ref. 6]) is in compliance with the SRS Nuclear Criticality Safety Program, which is implemented by WSRC-SCD-3 (Ref. 3), and was designed to comply with the requirements specified by DOE Order 5480.24 (Ref. 2). DOE has issued DOE Order 420.1 (Ref. 5) as a replacement for DOE Order 5480.24, and Westinghouse Savannah River Company (WSRC) is developing an implementation plan for DOE Order 420.1. Due to the similarity of requirements between the two Orders, the compliance status of the Receiving Basin for Offsite Fuels (RBOF) is expected to be comparable. The Order requires mandatory compliance with selected American National Standards Institute (ANSI)/American Nuclear Society (ANS) standards related to nuclear criticality safety. Of those standards, the following ANSI/ANS standards are normally applicable to the RBOF:

ANSI/ANS-8.1-1983	Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors	Ref. 11
ANSI/ANS-8.3-1986	Criticality Accident Alarm System	Ref. 12
ANSI/ANS-8.7-1975	Guide for Nuclear Criticality Safety in the Storage of Fissile Materials	Ref. 13
ANSI/ANS-8.15-1981	Nuclear Criticality Control of Special Actinide Elements	Ref. 14
ANSI/ANS-8.19-1984	Administrative Practices for Nuclear Criticality Safety	Ref. 15

DOE Order 5480.24 also specifies ANSI/ANS-8.5-1986 (Ref. 16), but borosilicate glass raschig rings are not used as criticality controls in the RBOF and thus the standard is not applicable. Although not specifically required by DOE Order 5480.24, ANSI/ANS-8.17-1984 (Ref. 17) can be considered applicable (particularly as guidance) to the RBOF since many of the fuel types received, stored, and handled in the RBOF are from reactor facilities using light water moderation.

Criticality safety requirements for the RBOF are specified in SRS documents, including the following:

WSRC-SCD-3	WSRC Nuclear Criticality Safety Manual (U)	Ref. 3
WSRC-TR-95-0302	Nuclear Criticality Safety and Fuel Receipt Manual - Spent Fuel Basin Facilities (U)	Ref. 6
WSRC-TS-95-11 (S-TSR-H-0002)	Technical Safety Requirements, Receiving Basin for Offsite Fuels Facility and Resin Regeneration Facility (U)	Ref. 18

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6.3 DESCRIPTION OF POTENTIAL CRITICALITY HAZARDS

6.3.1 DESCRIPTION OF STRUCTURES, SYSTEMS, AND COMPONENTS POTENTIALLY CONTAINING FISSIONABLE MATERIALS, AND THEIR OPERATION

The RBOF basins are located in Building 244-H. Fissionable material is handled and stored in the basins. The water-filled basins are constructed of reinforced concrete, with design water depths ranging from 22 to 29 feet. The basins were designed for the safe handling and storage of irradiated fuel from a variety of reactor facilities, including SRS facilities. The RBOF facilities include the provision for repackaging fuel for storage and/or shipment. Processing capabilities for all the fuel types in the RBOF are not available at SRS, and several fuel types have been in the RBOF nearly since RBOF storage was initiated. The storage arrangements (structures) in the RBOF are designed for significant flexibility in storage configurations. Irradiated fuel is handled and stored under water, generally with 5 to 10 feet of water above any fuel components.

The current mission of the RBOF includes the following:

- Receiving Department of Transportation (DOT)-approved casks from offsite
- Unloading those casks
- Packaging the fuel for storage in the RBOF and other storage facilities (e.g., L-Reactor Basin)
- Storing the fuel
- Shipping the fuel in SRS or offsite casks to those storage facilities
- Receiving and unloading onsite casks

The receipt, storage, and shipment of failed fuel may also be included.

Figure 6.3-1 provides a general plan view of the RBOF facility. Detailed descriptions of the facility structures are provided in Chapter 2. Upon receipt, fuel generally is unloaded from shipping casks in the cask basin, identified, transported to working basins for cropping of excess material (as appropriate) and insertion into storage bundles (or buckets), and transported (in bundles or buckets or as individual units) for placement in the storage racks. Some fuels, such as Material Test Reactor (MTR) fuels, upon removal from a cask, may be evaluated using a Research Reactor Fuel Counter (RRFC) to determine the fissile content of the fuel element.

Figure 6.3-2 provides a layout of the #1 and #2 storage basins in the RBOF. The storage racks provide a framework to establish parallel rows of fuel, with the nominal center-to-center spacing between rows defined by the aluminum I-beams comprising the storage structure. In the #1 storage basin, rows are available with spacings of 4, 6, 9, 11, 12, 16, and 25.5 inches. In the #2 storage basin, 12-inch rows are available. In addition, six modular racks, referred to as the

Expanded Basin Storage (EBS) racks, have been installed in the #2 storage basin and provide spacing of a minimum 9 inches across the rows. Hardware, described in Sections 2.4.1.3 and 6.3.1.3 and in EFR-RSE-95-0100 (Ref. 10), provides the positioning and spacing of fuel bundles within the non-EBS rows. Figure 6.3-3 provides a pictorial view of RBOF storage rack hardware arrangements.

Several types of storage bundles are in use. The bundle used for a specific fuel type generally depends on the cross-sectional fuel assembly dimensions. Table 6.3-1 lists characteristics of typical bundles.

The bundles most used are the square tube dissolver bundle, general purpose tube bundle, and EBS bundle. Depending on the size and reactivity of the fuel assemblies, generally two to six fuel assemblies may be placed in a bundle (most expected fuel receipts will involve MTR type fuel). In some cases, fuel may arrive already bundled and will be stored in the bundles. The bundles (or individual fuel assemblies or components) are then placed in the storage rows, with spacings between specified groupings of bundles (or components) that have been demonstrated by analyses to ensure criticality safety if the fuel bundles remain within the constraints of the rows. Some fuel may be placed in storage buckets, rather than in bundles, and placed in the bucket storage racks (Figure 6.3-2). Fuel that has failed, such as with breached cladding, is generally canned prior to storage or is stored in test tube bundles, which are placed in the test tube storage racks that provide more isolation from neighboring fuel than the standard storage racks.

Sections 2.3.4, 2.5.1, and 2.5.2 describe the fuel receipt, handling, storage, and shipping activities in the RBOF. Provisions are available to disassemble some fuel for inspections, as needed. An underwater saw is available for cropping fuel assemblies of excess nonfissionable material (end fittings). Facility-specific cask and fuel handling activities are described in either the 3000 or 4000 series Normal Operating Procedures (NOPs) for the RBOF. Fuels are received from offsite in a variety of shipping casks, approved in accordance with DOT regulations for shipment of irradiated fuel. Fuels are generally shipped between SRS facilities in the SRS 70-ton cask. WSRC-TR-95-0302 (Ref. 6) provides some description of RBOF activities and criticality safety considerations. Brief descriptions of the RBOF activities of criticality significance derived from the RBOF Double Contingency Analysis (Ref. 19) are incorporated in the following sections.

6.3.1.1 Cask Unloading

Casks containing spent fuel elements are delivered to the RBOF carport by truck or railroad car. A 100-ton overhead crane is used to lift and transfer the cask to the cask wash pit where preliminary unloading operations are performed prior to transfer to the cask basin, including removal of impact limitors and loosening and removal of most cask lid bolts. The cask is then moved to, and lowered into, the cask basin until the top of the lid is above the water surface to facilitate removal of the remaining lid bolts. The remaining lid bolts are removed, and the cask is lowered to the bottom of the basin. The lid is then removed and stored. Personnel work from

a traveling bridge crane and use tools to remove the basket spacers/lids from the cask to allow removal of the fuel. In some cases, the entire basket is removed.

Based on the NSDS limits for the fuel type being unloaded, the operating procedure is customized to specify the number of fuel elements to be placed in a transfer bucket. Operations, Criticality Safety, and Engineering provide review of each customized operating procedure prior to use. The NSDS specifies the maximum safe number of elements that may be handled together (i.e., placed in a single bucket). A transfer bucket is positioned on the cask basin floor away from the cask. An unloading bucket is positioned adjacent to the cask at the level of the cask basket. Tools designed for remote handling are used to remove a fuel assembly and to place it in the unloading bucket. The unloading bucket is then moved to a position on the cask basin floor near the transfer bucket. Unloading and transfer buckets are separated by an isolation bucket. Tools are used again to remove the assembly from the unloading bucket for identification. Following verification of identification, the fuel assembly is placed in the transfer bucket. Inserts are generally used in the transfer bucket to limit the number of fuel assemblies that can be loaded into the bucket. The empty unloading bucket is then returned to the position adjacent to the cask, and the process is repeated until the designated number of assemblies have been placed in the transfer bucket. The transfer bucket is then transported to a working or storage basin.

An RRFC (Ref. 20) has been installed in the RBOF cask basin. The intent is that sample assemblies from shipments containing MTR type fuel assemblies will be inserted, individually, into the RRFC to determine the amount of U-235 in the fuel assembly. That measurement is compared to the manifest records accompanying the shipment to confirm fuel characteristics.

In some situations, cask baskets are removed from the cask and placed on the cask basin floor prior to fuel unloading. Prior to removing the loaded cask basket, the basin is cleared of fuel. The cask basket is lifted and placed on the basin floor or basket stand, and fuel unit unloading is accomplished as described above. Administrative restrictions generally prohibit removal from casks of cask baskets loaded with more than the safe number of fuel pieces.

Criticality safety during cask unloading operations is maintained through administrative control of the mass of fissile material (number of fuel pieces) placed within the unloading and transfer buckets. Inserts may be placed in transfer buckets to limit the number of assemblies in a transfer bucket. Administrative control is also used to ensure that adequate spacing between loaded buckets is maintained.

Criticality concerns associated with the cask unloading operation arise from the following:

- The potential to receive fuel with characteristics outside the acceptable (analyzed) range
- The potential to drop fuel, thereby creating an unanalyzed (assumed unsafe) configuration
- Cask handling events (e.g., dropping a cask) leading to an unanalyzed (assumed unsafe) configuration

Analyses are performed for cask loading and unloading conditions to demonstrate subcriticality of possible configurations of dropped assemblies and cask baskets. Criticality concerns also arise from the potential to place an unsafe number of fuel elements into an unloading or transfer bucket. Violation of spacing controls between cask baskets, unloading buckets, and/or transfer buckets can also result in a criticality.

6.3.1.2 Fuel Bundling and Repackaging

Fuel packaging generally takes place in the repackaging basin where fuel assemblies are loaded into bundles, packaged, and transported to storage or cask loading. Some bundling activities may be performed in the cask basin. Based on NSDS limits for the fuel type being packaged, the operating procedure is customized to specify the number of fuel assemblies to be placed in a bundle. The NSDS specifies the safe number of assemblies that can be in an isolated uncontrolled configuration. No more than that number of assemblies may be placed in a single bundle. The physical limit for a bundle may be less than the safe number.

An empty bundle and lid are cleaned, assigned identification numbers, and placed on the tilt table. Using underwater hand tools, a fuel assembly is removed from the transfer basket and placed in the bundle. Additional assemblies are placed in the bundle until the specified number is reached. The lid is then placed on the bundle top and secured. Lids have a bail to facilitate handling.

Preparation of assemblies for bundling can require underwater cropping of inert ends from fuel assemblies. The cropping process employs a band saw attached to a special saw table in the repackaging basin. A fuel assembly is positioned on the table and held firmly against a stop while the saw is used to crop the end from the assembly. The scrap cropped end is removed from the table and placed into a scrap bucket.

Criticality safety in the repackaging and unbundling operations is maintained through administrative control of the mass of fissile material (number of fuel assemblies) placed within a fuel bundle. It is also maintained through administrative restrictions on fuel accumulation outside the confines of the transfer bucket or fuel bundle. No neutron absorbers are normally present during repackaging activities. Spacing between bundles and between bundles and transfer buckets is procedurally controlled.

Criticality concerns associated with the repackaging operation are related to fuel bundle assembly. It is possible, through procedural violation, to package an unsafe quantity or mixture of fuel types. In the case of empty bundle filling, errors could result in an excess quantity of material being placed in a bundle. For packaging a partially filled bundle, bundle identification and selection errors may occur in addition to specification and loading errors. Each of these failures could initiate criticality in the repackaging/bundling operation through violation of a mass limit (safe number limit). In general, the criticality concern for an overloaded bundle is failure of the bundle, resulting in an uncontrolled (optimal) configuration of greater than the safe number of fuel assemblies. Criticality concerns also arise from the potential for cropping errors

that would result in an unsafe quantity or configuration of fuel or the presence of fuel in unexpected places (no fuel is allowed in scrap buckets).

6.3.1.3 Fuel Storage

Intact fuel assemblies and bundled fuel elements and assemblies are normally stored vertically (near vertical) in rows within the #1 storage basin or the #2 storage basin. These rows are defined by a vertical framework of racks constructed of aluminum I-beams. Gratings, lugs, guide plates, and spacers, referred to collectively as "hardware," are installed between the racks to define the individual storage spaces along the rows. The hardware between the racks serves to both support and space the stored fuel. In addition, the EBS storage racks have been installed in the #2 storage basin to provide the capability to store additional fuel elements in the vertical position. EFR-RSE-95-0100 (Ref. 10) describes the RBOF racks and hardware. Table 6.3-2 is derived from EFR-RSE-95-0100 and provides dimensions of the hardware available for use in the RBOF non-EBS storage racks. Up to four tiers of this hardware (depending on the length of the stored fuel) can be installed in each row. Each tier consists of the following:

- A grating that supports and centers the fuel/bundle between the storage rack
- Two guide plates near the top of the fuel that center the fuel between the racks
- A number of spacers that bridge the guide plates to position the fuel along the row

Two types of spacers are employed:

- The swing-down spacer pivots on one guide plate to engage a slot in the other guide plate.
- The removable spacer fits into slots cut in both guide plates.

Rack ties bridge the I-beams defining a storage row at the top of the storage racks. These rack ties pivot on an I-beam and engage the adjacent I-beam to interlock and unite the whole rack system. Individual rack ties are raised to permit the movement of a fuel unit along the row and then are immediately re-engaged. A procedural requirement permits only one rack tie to be raised in a row at a time. A removable rack tie has been developed for use in situations where problems are encountered engaging installed rack ties. A lock installed at the end of each row bridges the gap between adjacent rack I-beams and prevents movement of fuel into or out of the row. The row lock can only be removed by a special tool that is under the control of the certified supervisor. It is sometimes necessary to store fuel in reactor slug buckets. A vertical storage frame similar to the main storage frames is positioned on one side of the #1 storage basin (Figure 6.3-2). If the integrity of the fuel cladding is suspect, the fuel is canned prior to storage, or is placed in an isolation container (referred to as a "test tube"), and stored in a special rack (test tube) near the door to the transfer canal. In both of these rack systems, safe spacing between fuel containers is ensured by hardware.

Temporary storage of fuel elements in transfer buckets, bundles, or test tubes is permitted in the inspection, disassembly, and repackaging basins and in the transfer canal. A loaded transfer bucket is placed on the floor at the end of a working basin. A combination of chains or chain(s) and isolation buckets are positioned across the basin, restricting access to the temporarily stored bucket and thereby ensuring that safe fuel spacing is maintained.

Criticality safety during fuel storage operations is maintained through control of fuel spacing. The rack and hardware systems provide passive-engineered control that, when supplemented by administrative control, ensure safe spacing between fuel containers stored in the #1 and #2 storage basins. Criticality safety during temporary storage within the disassembly and repackaging basins is ensured by administrative control alone. No neutron absorbers are normally present during storage activities.

Criticality concerns associated with fuel storage operations arise from the potential to violate NSDS requirements of fuel spacing and the potential for fuel rack displacement in the #1 or #2 storage basin if a chain hoist or fuel handling tool snagged a storage rack, displacing the fuel and resulting in an unsafe configuration. A criticality concern is also associated with the potential for corrosion (due to out-of-range basin chemistry) of bundles, fuel, or storage rack structures, resulting in a loss of spacing control or uncontrolled amounts of fuel debris on the basin floor.

6.3.1.4 Fuel Transfer (Within Basin)

Fuel units are moved within basins using the traveling bridge, mechanical hoists on the monorail system, hand tools, and manipulators. Fuel is transferred between basins through a system of canals. Fuel movement through the canal system is by means of an overhead monorail system.

Criticality safety during fuel transfer operations is maintained through isolation (1 foot water [TID-7016 (Ref. 21) indicates 300 mm of water is an essentially infinite thickness]) of fuel in transport from fuel in storage. Administrative controls require that fuel cannot be transferred with other fuel in process.

Criticality concerns associated with fuel transport operations arise from the potential to violate the administrative control of fuel spacing (isolation) and thereby violate safe number limits in groups of fuel. Introduction into the basin of materials that are better moderators or reflectors than water, and their subsequent transport in the vicinity of fuel, is also a criticality concern.

6.3.1.5 Cask Loading

Most shipments from the RBOF are to other onsite facilities. The 70-ton cask is used for those transfers. The process of loading a cask for shipment is essentially a reversal of the unloading process previously described. The quantity of fuel units to be loaded is determined based on NSDS restrictions. This quantity of fuel units is specified in the operating procedure and verified

through independent review. The quantity of fuel approved for loading may be limited by the capability for handling fuel at the receiving facility. Fuel to be loaded is identified and transferred from the repackaging or storage basin, via the transfer canals to the cask basin where cask loading operations occur. If required for the fuel type specified, slot blocking devices are positioned within the cask to prevent incorrect placement of fuel units. In some cases, neutron-absorbing material is positioned within the cask to reduce the potential for criticality. Fuel units (assemblies or bundles) are placed within the cask individually until the designated loading quantity has been reached. The cask lid is put in place, and the cask is lifted to the surface where the bolts are tightened. The cask is then transferred to the cask wash pit for survey and decontamination, if required, and then onto a truck or railroad car.

Criticality safety during cask loading operations is maintained through control of the mass of fissile material (numbers of assemblies or bundles) placed in the loading and transfer buckets or shipping cask. Administrative controls require that the safe quantity of fuel allowed in a cask with specific spacing (interaction) and neutron absorber (poison) engineered controls be determined prior to loading that fuel.

Criticality concerns associated with the cask loading operation arise from the potential to load the cask in a manner that would violate criticality safety requirements of fuel quantity, spacing, and presence of neutron absorbers. Concerns also arise from the potential to drop fuel or to damage the cask (e.g., dropping a cask) or cask internals, resulting in an unanalyzed configuration. Inadvertent draining of a cask can result in enhanced moderation within the cask, leading to criticality.

6.3.2 FISSIONABLE MATERIAL INVENTORIES, ENRICHMENTS, FORMS, AND LOCATIONS

Fissionable materials in the RBOF are generally in the form of irradiated fuel and target assemblies. Some fuel is present as fuel element pieces in shipping or storage canisters. Most fuel has been placed in aluminum storage bundles with two to six fuel assemblies per bundle. The number of fuel assemblies per bundle depends on the reactivity of the fuel assemblies. Only one type of fuel is in a given fuel container. Frequently, all the fuel on a single RBOF row will be from the same reactor facility. Each fuel type is identified by source facility, and that identification is maintained while the fuel resides in the RBOF.

Most fuel in the RBOF basins consists of enriched uranium. More than one-half of the fuel in the basins is highly enriched uranium fuel assemblies composed of thin aluminum-clad U-Al fuel plates, commonly referred to as MTR fuel. Low enrichment fuels and targets were generally irradiated to result in considerable Pu-239 buildup. Other fuels in the basins include enriched uranium cermet plates or rods clad in aluminum, steel, or zirconium; U-Zr fuel tubes clad in zirconium; and uranium oxide (normally as powder) in aluminum cans. Some fuel types were manufactured with uranium metal, uranium oxide, plutonium metal, or plutonium oxide as the fissile material.

Table 6.3-3 is derived from EPD-CTG-95-0026 and RRD-SRB-960002 (Ref. 22, 23) and indicates a typical inventory of the #1 storage basin (Rows A-S and AA-WW). Table 6.3-3 provides a detailed typical breakdown of fuel types and inventory information for the #1 storage basin as an example of the many varying types of fuel received, handled, and stored in the RBOF. The table indicates the fuel type, the applicable NSDS, the fuel form, the fissile loading of the assembly, the safe number (N_{safe}) of assemblies, the number and type of fuel assemblies (bundles) in each row, and other information. As fuel is received into or shipped from the RBOF, fuel may be relocated within the RBOF for more efficient storage. The inventory indicated by Table 6.3-3 should not be considered a static condition.

6.4 INADVERTENT CRITICALITY CONTROLS

Inadvertent criticality control is achieved with process designs and operations which ensure that criticality safety parameters are maintained within safe ranges. The three basic means of control are passive-engineered control, active-engineered control, and administrative control.

Passive-engineered control is preferred over active-engineered control, and active-engineered control is preferred over administrative control. WSRC-SCD-3 (Ref. 3) and Engineering Guide 13110-01-G (Ref. 24) provide details about each means of control. Chapter 6 of the Generic SAR (Ref. 4) provides a generic discussion of engineered and administrative (procedural) controls generally used at SRS. Only the controls specific to the RBOF are discussed in this SAR.

The RBOF uses a combination of passive-engineered and administrative controls to prevent the formation of potentially critical configurations of fissionable materials in the RBOF basins. Only one active-engineered control has been identified in the RBOF. Controls in the RBOF are directed towards prevention of uncontrolled configurations containing more than a safe fuel unit (e.g., assemblies, bundles of assemblies, cans of scrap, total mass within a storage unit or storage array) and towards prevention of storage array configurations with inadequate spacing between groupings of fuel. Other than cropping of excess nonfissionable material from the ends (end fittings) of some assemblies, no processing (e.g., changing material form, changing chemical composition, changing material geometry) of fissionable material takes place in the RBOF.

Some fuel assemblies may be disassembled for inspection, but the disassembled fuel is repackaged in accordance with limits specified in the applicable NSDS. Some fuels may be repackaged from their original shipping containers to SRS storage bundles, but the intrinsic fuel form is not changed. Correspondingly, some fuels may be repackaged (changing number of assemblies per bundle) when being shipped from the RBOF, but again the intrinsic fuel form is not changed. Fuels stored in the RBOF may be repackaged to maximize use of storage capacity (i.e., fuel consolidation) but only within analyzed approved limits specified in the NSDSs, but again the intrinsic fuel form is not changed.

6.4.1 ENGINEERED CONTROLS

The two means of engineered controls are defined as follows:

- Passive-engineered controls are fixed design features or devices that rely on natural forces or properties of material to control or prevent a nuclear criticality. These controls do not require human intervention or the activation or movement of a device by electrical, mechanical, hydraulic, or other means.
- Active-engineered controls are devices that require activation or movement by electrical, mechanical, hydraulic, or other means to control or prevent a nuclear criticality. These devices sense parameters important to criticality safety and

automatically initiate action to secure the system to a safe condition without human intervention.

As indicated in Chapter 6 of the Generic SAR (Ref. 4), engineered controls may include the following:

- Spacing (interaction) control (distances between individual fuel assemblies/bundles or arrays of fuel)
- Neutron absorber (poison) control (presence of neutron absorbers to reduce neutron interactions within equipment or between fuel in arrays)
- Concentration (density) control (restrictions on fissile content per unit volume, particularly in solutions)
- Moderation and reflection control (restrictions on the presence of moderators within arrays or external to arrays of fissionable material)
- Mass and volume control (restrictions on the mass or volume of fissionable material in a fissionable unit or array of fissionable units and can include control of the number of fissile units in a "safe fuel unit")
- Enrichment control (restrictions on fissile enrichment, such as U-235 content in uranium)
- Geometry control (dimensions and shapes of fissionable material)

In the RBOF, only spacing control is considered to be a passive-engineered control for fuel handling and storage. However, during receipt at the RBOF and shipping to SRS facilities, neutron absorber control is also considered a passive-engineered control.

The following sections delineate how spacing control and neutron absorber control are implemented as engineered controls in the RBOF and discuss why, with one exception, no other engineered controls are employed in the RBOF for fuel handling and storage.

6.4.1.1 Spacing (Interaction) Control

Spacing or interaction control involves the use of distance, arrangement, and shielding (neutronic isolation) restrictions between and among units, vessels, containers, equipment, and accumulations of fissionable materials to minimize the potential for neutron interaction of fissionable materials.

The application of spacing (interaction) controls and the factors used in the design of these controls involve the following items:

- Limitation on spacing between units, vessels, containers, and equipment containing fissionable materials that are stored or moved such that the entire array (or batch) will remain subcritical under credible conditions that might affect the array (or batch)

- Reliance on the integrity of storage racks to maintain spacing control during and following design basis accidents

Storage configurations and irradiated fuel handling in the RBOF occur under water. Thus, the control of spacing between fuel units or arrays of fuel units is very important, since the fuel is fully moderated and reflected in the storage arrays. Nearly all storage in the RBOF is in fuel storage racks, which arrays the fuel in parallel rows (Figure 6.3-2). Where fuel spacing is not controlled by the storage racks, administrative controls are imposed to ensure that fuel units are isolated from one another. For example, the structure of the bucket storage rows in the #1 storage basin is such that approved buckets properly positioned in the structures have at least 1 foot of water isolation on each side separating each bucket from adjacent buckets or other fuel storage units (e.g., bundles, tubes). Row sizes are determined by the horizontal distance between centerlines of adjacent I-beams. Those nominal centerline-to-centerline spacings may be 4, 6, 9, 11, 12, 16, or 25.5 inches (Figure 6.3-2). Fuel units placed in the row are positioned in the row via "hardware," which nominally centers the fuel unit on the centerline of the row and separates the fuel unit from the adjacent fuel units by the thickness of the spacer. The upper end of the fuel unit is fixed by the spacer hardware, and the lower end of the fuel unit is nominally positioned on the row centerline by the "trough" hardware on the support grating. The hardware is described in Section 2.4.1.3. Some description is also provided in Section 6.1.4 in the glossary terms and in Section 6.3.1.3. The use of the spacers provides an average spacing between fuel positions within the row. Table 6.3-2 provides dimensions of the hardware.

The #2 storage basin racks are a vertical framework of aluminum I-beams. The racks are spaced so that there are 11 rows with 12-inch center-to-center spacing. The effective lateral storage space in each row is 9 feet with the vertical height of the racks extending 18 feet 5.25 inches. One row of rack ties is provided to interlock the system. Hardware is specifically provided for the #2 storage basin racks and provides for up to 60 storage positions and two tiers.

Six modular racks, referred to as the EBS racks, have been installed in the #2 storage basin to provide the capability to store additional fuel elements in up to 180 vertical positions. Each rack consists of 30 aluminum tube sections (6 inches by 6 inches by 12 feet high) in which the fuel bundles are stored. The storage racks provide storage of fuel bundles at a nominal center-to-center separation distance of 9 inches across a row. The primary function of the tube sections is to maintain spacing requirements for criticality between the fuel bundles.

The RBOF storage row hardware does not form a tightly fitted matrix of fuel. There is some space between the upper hardware pieces that allows some movement of the fuel bundle relative to the hardware and the row centerlines. The possibility exists that fuel units may tilt towards each other (either at the top or bottom of the fuel unit) and either within the row or across rows. Across-row tilting is limited by the presence of the trough (into which the fuel unit is placed when positioned in the row) and the I-beams that form the row structure. On occasion, bundles have been noted to not be centered in the trough guides, but the I-beams generally prevent the tilting fuel bundles from significantly intruding on the adjacent row.

The degree of tilting possible was recently determined for each type of fuel and each type of fuel bundle in the #1 storage basin and was compared with evaluations of reactivity interactions between tilting fuel (Ref. 22). That evaluation determined that tilting of the fuel stored in accordance with the NSDS requirements, within the constraints of the row hardware and the row I-beams, would not result in exceeding reactivity limits in the RBOF. Appropriate revisions of the NSDSs were processed as a result of the tilting fuel analysis. Since then, the criticality safety analyses supporting each new or revised NSDS include evaluation of tilting of the fuel within the storage hardware. NSDSs (new and revised) and procedures now require the placement of an empty storage bundle between groups of fuel as they are placed into rack storage (non-EBS racks) to reduce the possible amount of in-row tilting.

The bucket storage rows are configured such that the swing-down gratings will provide 1 foot of water between Type A, B, or C storage buckets horizontally and vertically. When a Type D bucket is used in a lower tier, administrative controls on spacing are also necessary to provide a 1-foot isolation region. Buckets that may have fuel pieces extended above the bucket may have additional administrative restrictions, such as mass limits for groups of buckets, or required empty spaces (Nuclear Safety Control [NSC] spaces) above such buckets.

Analyses are performed for each fuel type to demonstrate criticality safety of that fuel in the storage arrays with the most reactive spacing possible. The criticality safety design limits on the spacing control, the bases that support their application, and the design criteria used to ensure subcritical configurations under all normal and abnormal conditions are briefly discussed in Section 6.5.4 describing analyses and evaluations. They are also noted in Section 6.4.3.1, which describes NSDS usage.

Reliance is placed on the integrity of storage racks to maintain spacing control during and following accidents. The accident analyses discussed in Chapter 3 address the risks and consequences associated with criticality accidents in the RBOF. Natural Phenomena Hazard (NPH) events (earthquakes, tornadoes, high winds, etc., disrupting building structures significantly) and events such as an overhead crane falling onto the storage racks would disrupt the storage arrays sufficiently to result in criticality. The racks are not expected to withstand all such events and remain intact. Operational errors (e.g., a traveling bridge hook snagging on the storage rack structure) pose a rack deformation concern, but equipment design, procedural requirements, and operator attentiveness are expected, for such instances, to minimize rack deformation and the associated possibility of inadvertent criticality.

6.4.1.2 Neutron Absorber (Poison) Control

Neutron poisons are not present in the RBOF storage arrays (except as they may be part of a fuel element as received). No credit is taken in the criticality analyses for the presence of any neutron poison (such as fission products) that may be an intrinsic part of a fuel assembly. No elements of the use of neutron absorbers as a passive-engineered control, as presented in Chapter 6 of the Generic SAR (Ref. 4), are used in the RBOF in the handling and storage of fuel, except as noted below.

The criticality safety of shipping casks received at the RBOF depends on the components of the casks and baskets within the casks. Most such baskets are made of stainless steel, which acts as a neutron absorber. Accident analyses (Ref. 25, 26, 27) show that assuming the stainless steel of such baskets is absent frequently results in calculated k_{eff} values exceeding RBOF limits. EPD-CTG-95-0117 (Ref. 28) recommended that RBOF management institute requirements to verify the presence of basket components before initiating unloading of such casks. Since the offsite casks are licensed by the Nuclear Regulatory Commission (NRC), or other competent authority, and the basket components are a required aspect to satisfy the license, no specific controls in addition to those already described in NOP-244-H-4000 (Ref. 29) were deemed necessary to comply with the EPD-CTG-95-0117 recommendations. NOP 244-H-4000 provides multiple verifications (prior to approval of fuel shipment and upon receipt of fuel shipment) that details of cask internals and fuel geometry (including basket geometry and material) in the "Cask Loading List" agree with the Certificate of Compliance (including referenced prints) for the cask.

Shipment of fuel in bundles from the RBOF to other SRS facilities generally will take place in the 70-ton cask, which may use cadmium separator plates between groups of fuel bundles to maintain criticality safety. Where specified on the NSDS, the supporting analysis that provides the limits on loading fuel into the 70-ton cask credits the presence of the cadmium plates. Thus, the presence of the cadmium plates is a passive-engineered control for shipment via the 70-ton cask. The presence of cadmium in the cadmium plates must be verified on a periodic basis. Fuel shipments requiring the use of these plates may not be loaded unless the presence of the cadmium plates has been verified within the specified period.

Row "SS" in the main storage basin has specialized storage racks designed exclusively for the storage of High Flux Isotope Reactor (HFIR) fuel elements. The storage posts consist of stainless steel, which was included in the analysis model, and as such is a credited design feature. No periodic verification is warranted for this structural material.

6.4.1.3 Engineered Controls Not Used in the RBOF

The following controls are not used as engineered controls in RBOF fuel handling and storage:

- Concentration (density) control
- Moderation and reflection control
- Mass and volume control
- Enrichment control
- Geometry control

The concentration (density) of the fissile content of fuel components is not controlled at the RBOF other than by the fact that no fuel processing takes place in the RBOF. Fuel components are maintained intact (as received) in the RBOF, except for occasional disassembly (e.g., removal of targets) of a fuel assembly for inspection. No fuel components (e.g., plates, rods, tubes) are

processed in any way so as to change the fuel concentration within a component. For some fuel assemblies, excess aluminum (end fittings) may be removed, but precautions are enforced to ensure that no intrusion of the fuel region of an element occurs. No elements of the use of concentration (density) control as a passive-engineered control, as presented in Chapter 6 of the Generic SAR (Ref. 4), are used in the RBOF.

Moderation and reflection are obviously uncontrolled in the RBOF in that fuel handling and storage takes place under water. Thus, full moderation and reflection by water are assumed for all analyses of storage arrays. Since the presence of water acts to provide some degree of isolation between rows and arrays of fuel, there are procedural controls to limit the reduction of water density (due to temperature or introduction of voids or materials that act as voids). No elements of the use of moderation and reflection control as a passive-engineered control, as presented in Chapter 6 of the Generic SAR (Ref. 4), are used in the RBOF. A control on reflection is administratively provided in that no moderators or reflectors more effective than water are allowed in the basin within 1 foot of any fuel unless an evaluation has demonstrated the criticality safety of the potential arrangement.

Mass and volume are not controlled by any engineered device in the RBOF. No processing of fuel occurs, and thus fuel components retain the mass and volume as upon receipt. Analyses assume that fuel is at the maximum fuel content (mass) specified for that fuel. Except as noted above (fuel assembly cropping and disassembly for inspection), fuel integrity is maintained. No elements of the use of mass and volume control as a passive-engineered control, as presented in Chapter 6 of the Generic SAR (Ref. 4), are used in the RBOF. Administrative mass controls are imposed by limits on numbers of assemblies, elements, cans, or bundles or mass allowed in isolated groupings. The use of inserts in transfer buckets to limit the number of fuel assemblies allowed in the bucket could be considered an engineered control to supplement the procedural limit on the safe number of assemblies.

Enrichment of fuel is not monitored or controlled at the RBOF since no fuel processing occurs to change the fuel enrichment of any fuel form from the enrichment upon receipt. Analyses assume that enriched uranium fuel is at the maximum enrichment specified for that fuel. In addition, for fuels in which plutonium production (during irradiation) increases the reactivity over the manufactured form, the plutonium buildup is considered during the analyses supporting definition of Criticality Safety Limits (CSLs). No elements of the use of enrichment control as a passive-engineered control, as presented in Chapter 6 of the Generic SAR (Ref. 4), are used in the RBOF. The enrichment in the material being received is verified through administrative controls using a document called an "Appendix A Agreement" as part of the shipping contract. These Appendix A Agreements are prepared by the shipper. Engineering verifies the Appendix A information to be within the parameter ranges bounded by the Nuclear Criticality Safety Evaluation (NCSE) analysis and indicated in the fuel description of the appropriate NSDS.

Geometry control involves (1) the use of dimension and shape restrictions on equipment to provide "geometrically safe" or "geometrically favorable" containers, vessels, drains, sumps, etc., for fissionable materials, or (2) restrictions on fluid flow preventing fissionable solutions from

assuming an unsafe geometry. The geometry of fissile units could be considered controlled at the RBOF in that all fissionable materials arrive at the RBOF as fuel assemblies, fuel pieces (or elements) in cans, or forms of scrap (including UO₂ powder) in cans, and those fuel forms are not changed during handling and storage in the RBOF. (Portions of end fittings [containing no fissionable material] of some fuel assemblies may be removed by cropping). In general, each of those individual fuel assemblies or fuel pieces (fuel units) is critically safe.

The dimensions and fuel contents of the fuel assemblies, elements, and materials, as specified in the Appendix A, are used in the criticality safety analyses determining the applicable CSLs for those fuel types. However, this maintenance of fuel form integrity is not considered an engineered geometry control. Many fuel types extend above the storage rack structure and thus cannot be considered to be protected from falling objects by the storage rack structure. No elements of the use of geometry control as a passive-engineered control, as presented in Chapter 6 of the Generic SAR (Ref. 4), are used in the RBOF. Use of buckets with inserts to limit both the number of assemblies and the assembly arrangement in a transfer bucket could be considered a form of geometry control, but the intent of the inserts is to limit the number of assemblies and not the shape of the array in the bucket.

Fuel integrity can be expected to be compromised in NPH incidents, due to the potentially large amount of facility damage. The Beyond Design Basis Event of storage pool draining (result of seismic event) with fuel melting and criticality is addressed in Section 3.4.3.

Specific analyses may be required to assess damaged fuel forms, whether as received or as damaged during handling in the RBOF. Damaged fuel forms are generally not stored in the same storage arrays as undamaged fuel. Some damaged or disassembled fuel, received in storage cans or canned at the RBOF, may be placed in normal storage arrays based on approved analyses.

6.4.1.4 Active-Engineered Controls Used in the RBOF

Active-engineered controls are devices that require activation or movement by electrical, mechanical, hydraulic, or other means to control or prevent a nuclear criticality. These devices sense parameters important to criticality safety and automatically initiate action to secure the system to a safe condition without human intervention.

The hazards analysis of Chapter 3 identified a potential criticality concern involving operation of the 100-ton crane. This crane is used in the RBOF to support cask handling and fuel handling activities. The crane operates over the cask vehicle unloading area, the yoke storage area, cask wash pit, and cask unloading basin. The current #2 storage basin was originally designed as a second cask unloading basin, and, as such, the crane rails allow for travel over the #2 storage basin. At the end of the crane rails, mechanical stops are provided to preclude over-travel of the crane itself.

The accident scenario involves the potential for disruption of the storage arrays should the crane impact the mechanical stops while loaded, followed by a loss of the load into the #2 storage

basin. Preventing the crane from traveling over the #2 storage basin and preventing crane travel toward the end rail stops effectively reduces the frequency of this scenario. Chapter 2 provides a more detailed description of this equipment.

Administrative controls (procedures) exclude operation of the crane over the #2 storage basin, but since the crane is electrically driven, an active-engineered control is provided as a second contingency. A south travel limit switch is installed such that electrical power to the crane motor is disrupted, allowing the fail-safe brakes to actuate, stopping the loaded crane before it contacts the mechanical stops.

This active-engineered control is protected by a Technical Safety Requirement (TSR) Limiting Condition for Operation (LCO), applicable in all modes, with associated surveillances to ensure operability. There are no other active-engineered controls in use at the RBOF.

6.4.2 ADMINISTRATIVE CONTROLS

Administrative controls require personnel interactions with the process to verify that the process safety parameters remain within specified limits. Administrative controls are generally steps in procedures (including surveillance programs) and may involve action, caution, or verification steps in a procedure, or steps in a surveillance program, that rely on the judgment, training, and responsibility of people for implementation. Thus, administrative controls are frequently referred to as "procedural controls." Procedural controls for criticality safety generally involve nuclear material safety limits (e.g., mass, presence of moderators, presence of neutron absorbers, changes in array spacings, and procedures for handling, storing, and transporting fissionable materials). Examples of such controls presented in Chapter 6 of the Generic SAR (Ref. 4) include the following:

- Sampling and reviewing sample results
- Recording and reviewing material transfers
- Calculating totals of material transfers
- Verifying material types and quantities introduced to a system

Procedural controls affecting criticality safety or related to criticality safety in the RBOF generally involve the following or similar items:

- Verifying the fuel type and characteristics of material to be received
- Verifying the adequacy of shipping casks (for receipt shipments) to ensure compliance with RBOF criticality safety criteria
- Handling all fuel within safe mass or safe number limits (groups of individual fuel units or bundles of fuel) throughout operations

- Maintaining required spacing between groups (safe mass or safe number) of fuel during transport or temporary storage in canals and working basins
- Packaging fuel and verifying that fuel is packaged within the requirements (fuel assemblies per bundle) specified in the NSDSs
- Placing the fuel in storage racks and verifying that the fuel is positioned in storage racks within the hardware specified in the NSDSs
- Placing the fuel in the storage racks and hardware at the spacing requirements specified in the NSDSs and verifying that the spacing requirements are met
- Placing the fuel in the storage racks only adjacent to other fuel types as specified in the NSDSs and verifying that the placement requirements are met
- Providing isolation between individual fuel assemblies following the removal from casks until the fuel is identified to be in agreement with shipping manifests and the Appendix A
- Allowing only one cask at a time in the cask basin
- Verifying that moderators, other than water, in the basins are controlled and not placed within 1 foot of storage arrays or groups of fuel being handled unless explicitly analyzed and approved by NSDS requirements or the Unreviewed Safety Question (USQ) process
- Incorporating the CSLs specified in the NSDSs into the operating procedures and verifying that incorporation is correctly performed
- Ensuring that basin chemistry is within limits specified in accordance with the TSR Program (basin chemistry is important to minimize corrosion of fuel and fuel containers in the basin)
- Verifying the presence (when required) of cadmium separator plates in the 70-ton cask before placing fuel in the cask
- Performing and verifying the performance of blackness testing of cadmium separator plates in the 70-ton cask to ensure the presence of neutron poison
- Verifying basin water level (primarily to ensure water shielding over the fuel to reduce radiation to workers during operations, but also to reduce the potential dose from a criticality accident under water consistent with the accident and risk analysis for the RBOF)
- Properly using bucket inserts, the HFIR blocking device, and other specified NCSE requirements

LCOs and associated surveillance requirements for the RBOF are specified in the RBOF TSR (Ref. 18) and were derived from the accident analysis of Chapter 3. The majority of these requirements were previously in place and based on the Basis for Interim Operation (Ref. 30) and Technical Standards for the RBOF (Ref. 31). Fuel-specific handling and storage limits are

developed as part of the NSDS Program, and other operational controls may also be designated based on the RBOF Double Contingency Analysis (Ref. 19) or specific NCSEs. Organizational, programmatic, technical, and management controls (referred to as administrative controls) are also specified in those documents.

The surveillance requirements that comprise implementation of those TSR controls related to criticality safety are summarized in Table 6.4-1. Table 6.4-1 is intended to provide examples of the TSR requirements and does not necessarily represent the latest version of those requirements in the latest effective revisions of the TSR. Those controls and surveillances are implemented in the RBOF 1000, 3000, and 4000 series NOPs. Specific criticality safety rules for RBOF operations are provided in a "reference" NOP, NOP 244-H-4818 (Ref. 32). This procedure is used in pre-job briefings to remind personnel of the criticality considerations and limitations applicable to the planned work. This procedure supports the specific implementing procedure for the work activity. NOP 244-H-4000 (Ref. 29) proceduralizes the process of verifying that fuel shipments proposed for receipt at the RBOF have been evaluated to ensure that the fuel can be safely received and handled in compliance with the requirements specified in applicable NSDSs.

Section 6.5 discusses administrative controls for the review and approval of changes to process or system configurations.

6.4.3 CRITICALITY CONTROL IMPLEMENTATION

WSRC-TR-95-0302 (Ref. 6) outlines the implementation of administrative and procedural criticality controls in the RBOF. Many of the procedural controls are necessary to ensure that the passive-engineered controls previously discussed are properly used.

There is no practical method to specify in the RBOF TSR (Ref. 18) a reasonably sized set of CSLs applicable as a whole to RBOF fuel handling and storage activities. Instead, the TSR addresses the administrative process of determining the CSLs on safe number and safe storage array configurations (spacings) for each fuel type, their specification in an NSDS, compliance with those limits, and verification of compliance with those limits. The NSDSs are appended to WSRC-TR-95-0302 (Ref. 6), which has a controlled distribution. As NSDSs are updated (revised), the approved revisions (provided by Regulatory Programs of the Spent Fuel Storage Division [SFSD]) are distributed by Document and Manual Control to holders of the controlled copies. Surveillances associated with CSL compliance are provided in the TSR and are directly implemented in RBOF procedures.

6.4.3.1 Nuclear Safety Data Sheets

The NSDS is the base document for nuclear criticality safety controls at the RBOF facility. Fuel handling evolutions that take place at the RBOF are governed by NSDS criteria specific to the type of fuel being handled. Figure 6.4-1 shows a typical NSDS.

NSDSs are developed by the Criticality Safety Engineering Group (CSEG) to document the results of calculations and judgments used in determining limits which ensure that nuclear criticality safety is maintained. An NSDS for a specific fuel type identifies the handling and storage requirements for that fuel type, including the following applicable items:

- Identification of source reactor
- Fuel unit type or class
- Fuel description, including dimensions, construction, initial fissile loading, and RBOF storage method
- Specific storage criteria, including the following applicable items:
 - Maximum allowable number of fuel units in worst configuration (safe fuel unit)
 - Within-row storage requirements (safe fuel group)
 - Across-row and along-row spacing criteria and required storage hardware
 - Other fuels involved in establishing across-row and along-row storage criteria
- Storage criteria related to the use of any additional constraints determined necessary (e.g., empty bundles)
- For some fuel types, the total U-235 mass allowed within an isolated grouping where isolation is provided by 1 foot of water or more between surfaces of fuel
- Data references (e.g., Appendix A, NCSEs, specific calculation records, related documents)
- SRS 70-ton shipping cask requirements

Analyses used in NSDS development are based on fuel specifications provided to the Department of Energy Savannah River Operations Office (DOE-SR) by the shipper in the form of an Appendix A. Upon receipt of an Appendix A, the CSEG performs an evaluation to determine if the fuel described is bounded by a current NSDS or if a new NSDS or NSDS revision must be developed.

A current, approved NSDS should be in place prior to DOE-SR approval for shipment of a fuel to the RBOF. A current, approved NSDS shall be in place prior to cask and fuel handling activities.

6.4.3.2 Procedures

The TSR requirements are procedurally implemented by the RBOF 1000, 3000, and 4000 series NOPs. Table 6.4-2 provides a representative (non-inclusive) listing of RBOF procedures for activities for which criticality concerns may exist. Table 6.4-2 is an example only to show the extent of procedures used in the RBOF; procedures listed there may be revised or deleted, or procedures may be added as operational needs arise or change. RBOF Procedures Management

maintains an updated index of procedures, cross-indexed to the TSR through a surveillance tracking database (see NOP 244-H-1607 [Ref. 33] as listed in Table 6.4-2).

Applicable NSDS/NCSE fuel handling and storage criteria are entered into the RBOF NOPs or Special Procedures by RBOF operations personnel prior to fuel handling evolutions. The information entered into the procedures is reviewed and approved for use by RBOF Operations, Engineering, and a representative of the CSEG. RBOF operations personnel performing fuel handling procedures document completion of NSC steps and prerequisites by signing or initialing after the action.

6.4.3.3 Pre-Receipt Fuel Evaluation

Prior to DOE-SR approval for shipment to SRS, specifications in the form of an Appendix A and associated irradiation history are evaluated by RBOF Operations, Engineering, and the CSEG. This evaluation is performed to ensure that the requirements of the RBOF Fuel Receipt and Shipping Program are met.

Applicable RBOF NOPs are prepared consistent with DOE requirements governing fuel receipt and shipping and are approved, maintained, and adhered to for all related operations. Programmatic and shipment records are retained in accordance with the SRS quality assurance requirements.

The Fuel Receipt and Shipping Program (see WSRC-TR-95-0302 and RBOF TSR [Ref. 6, 18]) ensures that casks received into the RBOF are either in compliance with the cask licenses as issued by the NRC and/or DOT/International Atomic Energy Agency (DOT/IAEA) or authorized for use in accordance with site procedures (Ref. 34) for onsite casks.

Engineering performs and documents an evaluation to determine if the cask/fuel meets the authorization basis assumptions regarding the following:

- Latest Safety Analysis Report for Packaging and Certificate of Compliance
- Fuel isotopic characteristics
- Maximum radionuclide inventory for an individual fuel element
- Maximum facility radionuclide inventory
- Maximum heat load
- Minimum cooling time
- Chemical compatibility (e.g., sodium content)
- Minimum water depth above active fuel (when storage position is identified, which may occur after shipment is authorized)
- Fuel weight (as related to the seismic analysis assumptions for the storage racks, which may also occur after shipment is authorized)

Fuel shipments not meeting these assumptions are evaluated by RBOF Operations, Engineering, and the CSEG using the USQ process (see Procedure 1.05 of WSRC Procedure Manual 11Q [Ref. 35]) prior to facility receipt. NOP 244-H-4000 (Ref. 29) is used by RBOF Operations as a checklist to ensure that the proper reviews have been completed.

6.4.3.4 Fuel Storage/Fuel Handling

Storage locations for incoming fuels are determined by RBOF Operations based on the requirements of the applicable NSDS and storage space availability. RBOF Operations requests the concurrence of Engineering and the CSEG with the proposed storage location. Across-row and along-row interactions involving the proposed storage location are detailed in the request.

If the proposed storage location is not bounded by current NSDSs, a review is performed by the CSEG to determine if an NCSE and/or NSDS revision is required or if a different storage location should be proposed. The proposed storage location is also evaluated by Engineering for compliance with other technical requirements (e.g., minimum water depth, tilt interactions) if not already verified.

Fuel movements to accomplish authorized storage are restricted to the appropriate safe fuel unit authorized by the NSDS. No more than one safe fuel unit is handled at a time in any one basin.

6.4.4 APPLICATION OF DOUBLE CONTINGENCY PRINCIPLE

The double contingency principle, as stated in DOE Order 5480.24 (Ref. 2) and in Section 6.4.3 of the Generic SAR (Ref. 4), is intended to demonstrate that no single failure (process upset) may result in a criticality accident. Independence and diversity of controls on the parameters (e.g., geometry, spacing, mass, volume, moderation, reflection, enrichment, concentration [density], and temperature) that can affect criticality are desired. Most desirable are independent controls on different parameters.

A double contingency discussion is required in each NCSE written in accordance with the requirements of WSRC-SCD-3 (Ref. 3). The double contingency discussion in each NCSE is intended to evaluate double contingency aspects of the proposed process, system, or equipment change, where appropriate. Thus, the double contingency status of the RBOF facility should be maintained up to date in the safety documentation for the facility. NCSEs (or equivalent documentation) prepared prior to WSRC-SCD-3 may not have discussion of double contingency aspects in detail. A comprehensive Double Contingency Analysis (Ref. 19) for current RBOF conditions and operations has been prepared in compliance with WSRC-SCD-3. Future RBOF conditions are covered under NCSEs prepared in compliance with WSRC-SCD-3.

6.4.4.1 General Summary of Double Contingency Compliance in the RBOF

The criticality concern in RBOF operations is generally confined to fuel handling and storage activities. There are no intended chemical or mechanical processes that can change fuel forms or in which process upsets can cause excess quantities, concentrations, or isotopic distributions of fissile materials to occur. Fuel is generally in solid, metal, or oxide forms, and no fissile solutions are involved. Since no chemical processing occurs, there can be no errors in chemical additions to result in process upset. Excess material (end fittings) may be cropped (sawed) from the ends of some fuel, such as MTR fuel assemblies. Errors in placement of the fuel on the saw table could result in exposure of fissionable material, scrap material containing unknown quantities of fissionable material in the scrap bucket, and unknown accumulations of fissionable material fines in the vicinity of the saw. Procedural and mechanical controls on the placement of fuel on the saw table are intended to prevent such errors. Corrosion in the basin is minimized by a Basin Water Chemistry Control Program, and corrosion (chemical change) in the fuel components (except in previously damaged and corroded items) is minimal.

Fuel handling in the RBOF basins takes place in the presence of water moderation and reflection. Since mass, neutron absorber, and spacing controls are used to maintain adequate margins of subcriticality, any mechanisms that can cause two controls to fail could plausibly lead to a criticality accident in the RBOF. Initiating events considered for double contingency analyses include those operational events that challenge the mass, neutron absorber, or spacing controls. NPH events (e.g., earthquake, tornado, and high wind) are also potential criticality initiators. Beyond Design Basis Events, such as an earthquake resulting in storage pool draining with resultant fuel melting (and criticality), are not considered, although this event is discussed in Section 3.4.3 for accident and risk analysis.

Controls in use in the RBOF (see Sections 6.4.1, 6.4.2, and 6.4.3) are inextricably linked with procedural controls (the weakest and least desirable of control mechanisms) even though the primary control in the RBOF is the physical spacing provided by the storage racks. The placement of fuel in the racks requires that operators place specific hardware in specific positions, directly control placement of fuel in its specified position, and then place other specific hardware in position to establish the specified spacing within the storage arrays. Sometimes the controls on spacing are not provided by hardware (racks) but are directly implemented by the operators placing and maintaining safe fuel units at least 1 foot from other fuel. Thus, emphasis is placed on proper attention to criticality controls by the operations staff. NOP 244-H-4818 (Ref. 32) is reviewed with the operations staff before each job to ensure that criticality considerations are clearly understood. Specific NSC steps in operations procedures are clearly marked to ensure that operators and supervisors are aware of the importance of the steps.

An RBOF team performing fuel handling activities consists of a minimum of an operator and a supervisor, providing immediate and somewhat independent confirmation that the activities are performed in accordance with the procedural requirements (the operator performs the activity and the supervisor provides surveillance that the activity is performed as specified in the procedure). Storage arrangements and operations activities are periodically confirmed through performance

of TSR surveillance requirements. Independent review functions provided by the SFSD Criticality Safety Committee also strengthen criticality safety of the fuel handling and storage processes.

CSLs are rigorously evaluated by the CSEG in NCSEs before they are included in NSDSs. The CSEG, Engineering, RBOF management, and DOE personnel review and approve NSDSs (see Section 6.5.4.2). Incorporation of CSLs in procedures is closely reviewed (and approved) by the CSEG, Engineering, and RBOF technical and management personnel to ensure that limits are properly incorporated. TSR surveillance requirements have been reviewed and evaluated and are under regular review to ensure that they incorporate all the surveillances necessary to ensure criticality safety in the RBOF. The TSR requirements are implemented in RBOF procedures. The surveillances include periodic reviews of storage conditions to ensure that storage conditions conform to the NSDS requirements.

No Common Mode Failure (CMF) aspects due to common power supplies, detection devices, alarm devices, etc., were identified by the RBOF Double Contingency Analysis (Ref. 19) or subsequent NCSEs performed to date. However, the fact that operations personnel are involved even in the passive-engineered control of spacing means that multiple operator errors (both operator and supervisor do not correct an error) could result in failure of the spacing controls (e.g., wrong hardware, hardware not positioned as specified, fuel not grouped in accordance with NSDS, fuel in wrong row, cask incorrectly loaded or unloaded). Inadequate testing and surveillance by personnel can result in spacers and poison plates improperly installed in shipping casks. This potential for CMF is mitigated by the emphasis on compliance with CSLs and attention to detail as required by conduct of operations. In addition, the periodic TSR surveillances provide an additional verification of correct fuel handling and storage.

A CMF that has been postulated for the RBOF facility is administrative in nature and involves the Appendix A supplied with the contract between DOE and the source facility for offsite fuels. Since the contract is a legal agreement between DOE and the shipper, the potential for the Appendix A to be inaccurate is extremely remote. Controls in the TSR Fuel Receipt and Shipping Program are designed to ensure the accuracy of the information. The controls mandate extensive checks to ensure that the data, upon which the NSDS limits are based, are correct. Any known discrepancies in the data are identified in the NCSE, which analyzes the significance of the discrepancies. Furthermore, all operators are trained to isolate fuel that does not match the description provided by the shipper.

6.4.4.2 RBOF Double Contingency Analysis

A comprehensive Double Contingency Analysis (Ref. 19) has been performed and documented for the RBOF and is summarized in this section.

TERMINOLOGY USED IN RBOF DOUBLE CONTINGENCY ANALYSIS

The following terminology was used in the RBOF Double Contingency Analysis (Ref. 19):

- **Administrative Control** - The least preferred means of control, which relies on the judgment, training, and responsibility of people for implementation. Administrative means of control may be action, caution, or verification steps in a procedure, or steps in a surveillance program.
- **Active-Engineered Control** - A means of control involving add-on, active hardware (i.e., electrical, mechanical, hydraulic) that protects against criticality. These devices act by sensing a change in the controlled parameter and then by providing automatic action to stop or reverse the undesirable parameter trend or otherwise secure the system to a safe condition without the necessity of human interaction.
- **Bounding Assumptions** - Values used for physical process parameters that are not directly controlled but represent the most reactive values that are credible. There are nine physical process parameters that affect neutron multiplication: fissile mass, moderation, geometry, spacing, density, neutron absorption, enrichment, reflection, and temperature.
- **Common Mode Failure (CMF)** - A CMF occurs when an initiating event can cause elements of two or more defenses to simultaneously fail.
- **Criticality Safety Controls (CSCs)** - The means by which physical process parameters are controlled within the CSLs. The preferred criticality control hierarchy for control reliance is passive-engineered, active-engineered, and administrative controls.
- **Criticality Safety Limits (CSLs)** - Values within which physical process parameters are controlled to ensure specified subcritical margins.
- **Defense** - A combination of bounding assumptions and CSLs (and associated CSCs) for the nine physical process parameters that, as a set, are sufficient to ensure the minimum subcritical margin against an initiating event. Failure of a defense results in a contingency.
- **Defense Element** - The individual values of the bounding assumptions, CSLs, and CSCs that make up the defense against an initiating event.
- **General-Diverse Control** - The most preferred criticality control method that satisfies the double contingency principle where CSCs are used to maintain the established CSLs for each of two or more independent physical process parameters. In this case, there are no defense elements in common, and hence CMF is minimized.
- **Initiating Event** - Process upset or other phenomenon that challenges any one of the bounding assumptions, CSLs, or CSCs. An initiating event consists of the change in condition and the resulting challenge to one or more of the provided defenses.

- **Passive-Engineered Control** - The most preferred means of criticality control, involving fixed passive design features or devices rather than moving parts. Independent of human interaction.
- **Physical Process Parameters (Parameters)** - Those parameters that directly affect the neutron multiplication of a fissile or fissionable system. Common usage is to group these in terms of the general components "mass, moderation, geometry." However, for the purpose of performing a comprehensive criticality analysis, it is important that the nine parameters remain distinct: fissile mass, moderation, geometry, spacing, density, neutron absorption, enrichment, reflection, and temperature. Acceptable values or ranges of values for these parameters are defined by either bounding assumptions or CSLs (maintained by CSCs).
- **Primary Contingency** - An unlikely, but credible, defense failure that results in an adverse change in the value of a physical process parameter for which CSLs have been established.
- **Redundant-Diverse Control** - A criticality control method that satisfies the double contingency principle, where CSCs used to control one physical process parameter are not only different, but also are as functionally different as practical to minimize the potential for CMF.
- **Redundant Like-Kind Control** - The least preferred (but acceptable) criticality control method that satisfies the double contingency principle, where CSCs rely upon separate, but functionally identical, mechanisms to control one physical process parameter.
- **Secondary Contingency** - An unlikely, but credible, defense failure that challenges or fails a CSC but does not adversely affect the value of a physical process parameter for which CSLs have been established.

RBOF DOUBLE CONTINGENCY ANALYSIS METHODOLOGY

A multi-step methodology was employed in the RBOF Double Contingency Analysis (Ref. 19):

- Fissile and fissionable material handling and storage were described for each category of operational activities (see Sections 6.3.1.1 through 6.3.1.5 for descriptions) that pose a potential criticality concern.
- Through evaluation of process and facility documentation, as well as operational requirements and practices, scenarios that could result in criticality were postulated.
- A logic model was employed that illustrated the various combinations of initiators and control failures that comprise each criticality scenario.
- Results from the logic model evaluation were examined in detail. The types of independent controls satisfying the double contingency principle required to prevent

violations or challenges of bounding assumptions or CSLs were identified for each postulated criticality scenario.

- Each criticality scenario was reviewed in detail, and the specific independent controls currently in place in the RBOF to prevent violations or challenges of bounding assumptions or CSLs for that scenario were identified. The additional controls required in the RBOF to ensure compliance with the double contingency principle were identified.
- The Double Contingency Analysis documentation provides descriptions of RBOF operations, the CSLs implemented in the RBOF, and the controls used in the RBOF. The analysis also provides discussions of each criticality scenario, indicating the strengths and weaknesses of the controls in place and the additional controls needed to provide a complete double contingency defense for that criticality scenario. Where CMF between independent controls is a concern, mitigating actions and factors were indicated. Summary tables of additional controls to be implemented and summary tables of the criticality scenarios are also provided.

CSLs for each fuel type handled in the RBOF are specified in the NSDSs appended to WSRC-TR-95-0302 (Ref. 6). CSCs are combinations of administrative mass and spacing controls and neutron absorber control. Mass control is achieved by limiting the number of fuel units placed in buckets and bundles, or being handled in a region, to the "safe number" specified in the NSDS. Spacing control is provided by proper use of the mechanical hardware associated with the storage matrix. Spacing is controlled administratively during fuel transport activities. Neutron absorber control is administratively ensured by insertion and testing of poison plates in the specified casks. Controls are addressed in the NOPS.

CRITICALITY ACCIDENT SCENARIOS, INITIATING EVENTS, AND FAILURE COMBINATIONS

The detailed hazards assessment for the RBOF (Ref. 36) included a systematic identification of those situations that could credibly pose a risk to criticality safety. Those situations formed the basis of the criticality scenarios used in the Double Contingency Analysis (Ref. 19) and are summarized in Table 6.4-3. The hazards assessment for the RBOF served as input for the development of a double contingency analysis to identify the failure combinations that could result in an RBOF criticality event and to identify the defenses required to comply with the double contingency principle as stated in DOE Order 5480.24 (Ref. 2).

The evaluation of the operations at the RBOF resulted in a list of failure combinations that could lead to criticality events. Each combination included an initiating event and one or more enabling events. The initiating event typically represents failure of a criticality control but may instead identify a change in an uncontrolled parameter. Each enabling event represents failure of a control. Verification that at least two independent controls (lines of defense) exist for each failure combination illustrates compliance with the double contingency requirement of DOE

Order 5480.24. Table 6.4-4 lists the criticality scenarios and the challenged bounding assumption or CSL identified in the Double Contingency Analysis.

An individual scenario is comprised of an initiating event and one or more enabling events. An initiating event may be direct failure of a control, a random mechanical failure, or an external initiator unrelated to engineered controls or administrative criticality controls such as NPH events. For the RBOF facility, external initiators include earthquake, tornado, and high wind. An enabling event may be failure of either a primary contingency or a secondary contingency. In some cases, multiple enabling events must occur for criticality to result. Details are provided in the Double Contingency Analysis.

COMPLIANCE OF THE RBOF WITH DOUBLE CONTINGENCY PRINCIPLE

The Double Contingency Analysis (Ref. 19) identifies controls which will ensure that the RBOF is in compliance with the double contingency principle from a perspective of operationally initiated criticality. In general, RBOF procedures specify operator actions and verification of conditions to ensure that parameters important to criticality are maintained within the appropriate CSLs during all fuel handling and storage activities or other activities that may affect fuel handling and storage. In a number of cases, the Double Contingency Analysis process revealed a need for additional independent verification of conditions consistent with the CSL requirements. The RBOF operating crew provides those additional verifications but not always in a formal proceduralized manner.

The major recommendation of the Double Contingency Analysis was to provide more procedural steps for explicit peer or independent verification of handling and storage conditions, such as the following:

- Maintaining isolation between fuel in transport and other fuel
- Ensuring no more than a safe number of assemblies in an isolated group or transfer bucket
- Verifying that correct hardware is installed in a storage row prior to initiating storage activities

The identified controls (or their equivalents) that were judged to be significant have been incorporated into the operating procedures. Suggested controls are being incorporated as procedures are being revised.

The Double Contingency Analysis indicates that the RBOF is not compliant with respect to NPH-initiated criticality because RBOF structures cannot be shown to prevent all structural damages that can impact the storage racks and result in criticality. Chapter 3 indicates that the overall risk of an NPH-initiated criticality is small.

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6.5 CRITICALITY PROTECTION PROGRAM

Nuclear criticality safety is administered at the RBOF by assigning responsibilities for key nuclear criticality safety requirements and activities to appropriate WSRC organizational units within and matrixed to SFSD. The management, organizational, and institutional safety provisions are discussed in Chapter 17.

6.5.1 CRITICALITY SAFETY ORGANIZATION

Direct responsibilities for the implementation of the site Nuclear Criticality Safety Program specified in WSRC-SCD-3 (Ref. 3), with respect to the RBOF, range from those required of the WSRC President to those required of the RBOF operators and auxiliary personnel. Site-level responsibilities for the WSRC President and staff organizations are discussed in Chapter 6 of the Generic SAR (Ref. 4) and are not detailed here. Administrative and organizational details of the RBOF-specific criticality safety organization and implementation of the RBOF Nuclear Criticality Safety Program are summarized in the following discussion. Figure 6.5-1 provides a representation of the organizational structure for criticality safety in the RBOF. Specific management group names and reporting structures may vary as organizations reorganize.

The full organizational structure of SFSD is not indicated in Figure 6.5-1. SFSD also includes groups who have responsibilities for implementing or assisting facilities in implementing management programs such as configuration control. Those groups may be called upon to provide specific services in fulfilling the management responsibilities of the facility management. The specific updated SFSD organization is provided in SFSD organization charts and is also available in the SRS SHRINE information system. The dashed lines in Figure 6.5-1 indicate that functions of Environmental, Safety, Health, and Quality Assurance (ESH&QA); the Criticality and Radiation Transport (CRT) Section of the Projects, Engineering, and Construction Division (PECD); and Emergency Preparedness are provided on a matrixed or interfaced basis. The dotted lines indicate matrixed or interface functions performed within SFSD.

6.5.1.1 SFSD General Manager and/or Chief Engineer Authority and Responsibility

The SFSD General Manager and/or Chief Engineer has authority and responsibility over the activities involved with ensuring criticality safety in the RBOF, as indicated for all SRS facilities in WSRC-SCD-3, Chapter 6 of the Generic SAR, and the RBOF TSR (Ref. 3, 4, 18), including the following:

- Accept and implement criticality safety responsibilities for the RBOF as delegated by the WSRC President
- Accept the responsibility for criticality safety for facility operations in the RBOF

- Ensure that applicable criticality safety standards and DOE Orders are applied in the design, modification, and operation of RBOF facilities
- Ensure that the WSRC policy for nuclear criticality safety, as stated in MP 4.5 of WSRC-1-01 (Ref. 1) and as implemented in WSRC-SCD-3, is applied at the RBOF
- Ensure that an interface or matrix arrangement exists with the CRT Section of PECD to provide necessary criticality safety expertise to RBOF activities and operations
- Ensure that appropriate training is developed and provided to the RBOF staff in compliance with applicable DOE Orders
- Delegate criticality safety responsibilities for RBOF activities to lower-level managers within the division as necessary, but maintain overall accountability for such responsibilities at the general manager level
- Establish and maintain a division/area-level Criticality Safety Committee, whose area of review and assessment includes the RBOF, and document the authority and responsibility of the committee in division/area manuals and/or committee charters (see Sections 2.2.8 and 13.1 of WSRC-SCD-3)
- Ensure that RBOF facilities are safely operated within the authorization basis and within environmental and safety guidelines
- Ensure effective interface with DOE Headquarters, DOE-SR, and the Defense Nuclear Facilities Safety Board
- Ensure that RBOF facility hardware improvements, the maintenance management system, procedural upgrades, and the Training Program are effectively implemented
- Ensure that SFSD departments are appropriately staffed and organized to support operation and maintenance of the RBOF

The Manager, SFSD Areas, reports to the Vice President and Manager, SFSD, and is responsible for the following:

- Operating RBOF facilities in accordance with the authorization basis and maintaining the RBOF in a safe condition at all times
- Controlling special nuclear material in the RBOF
- Ensuring that the RBOF operations staff is trained, qualified, and certified to safely operate the RBOF

The Manager, Regulatory Programs, reports to the Manager, Engineering (SFSD Chief Engineer), and is responsible for the following:

- Transmitting WSRC-approved NSDSs to DOE-SR for review and concurrence
- Distributing approved NSDSs

WSRC Procedure Manual RD-1 (Ref. 37) does the following:

- Provides the overall direction for the administrative and management systems implemented within SFSD
- Establishes linkage with WSRC administrative requirements
- Defines the organizational structure of the division
- Establishes the procedural hierarchy for the division
- Defines the administrative systems for organizations within the division

Table 6.5-1 lists and briefly describes the procedures in Manual RD-1 that have criticality significance for the RBOF. Table 6.5-1 is provided as an example of the number and types of administrative procedures provided within SFSD and is not intended to be a requirements list. The procedures in Manual RD-1 are frequently revised or may be superseded by other division- or site-level procedures. Chapter 12 discusses procedures and training for the RBOF organization.

RBOF MANAGER

Responsibility for RBOF management generally has been delegated from the SFSD General Manager via the Reactor Facilities Manager to the Basin Facilities Manager (RBOF Manager), who in turn has organizationally delegated technical responsibilities to the Deputy Basin Manager and operational responsibilities to the RBOF Operations Manager, as indicated in Figure 6.5-1. Some administrative and training activities may be coordinated with the Fuel Handling Manager. The general facility management responsibilities, as indicated in WSRC-SCD-3, Chapter 6 of the Generic SAR, and the RBOF TSR (Ref. 3, 4, 18), include the following:

- Accept responsibility for the day-to-day criticality safety in the RBOF. Specific activities may be delegated to a designee, but overall responsibility rests with the RBOF Manager.
- Accept responsibility for overall RBOF facility operation (including criticality safety) and delegate in writing the succession of that responsibility during any absence. The RBOF Manager shall have control over all activities necessary for safe operation of the RBOF.
- Accept responsibility and administer a program for Nuclear Incident Monitor (NIM) system management within the RBOF
- Accept responsibility for preparation, use, and maintenance of, as well as adherence to, facility procedures and drawings related to or affecting criticality safety at the RBOF

- Maintain a program of staff training in both general and facility-specific aspects of criticality safety at the RBOF, including qualification of fuel handlers
- Maintain a program of NIM familiarization and evacuation instruction for non-WSRC visitors and WSRC personnel needing access, but not normally assigned, to the RBOF
- Provide any additional criticality training deemed necessary for non-WSRC visitors and WSRC personnel needing access, but not normally assigned, to the RBOF
- Provide appropriately trained staff to determine when procedures, drawings, and design documents applicable to the RBOF require criticality safety review, ensuring that such procedures, drawings, and design documents are forwarded to the PECD CSEG for review
- Ensure that criticality safety is included in the RBOF self-assessments and that related deficiencies, action items, corrective actions, and other findings are properly addressed
- Accept responsibility for compliance with applicable DOE Orders, WSRC requirements, and authorization basis documentation applicable to the RBOF
- Approve the response to action items noted by the SFSD Criticality Safety Committee
- Ensure that criticality safety aspects of facility design, construction, and operation are covered by a documented safety analysis applicable to the RBOF
- Ensure that the RBOF is covered by a documented double contingency analysis as specified by the Nuclear Criticality Safety Program
- Accept responsibility for placing the RBOF in a safe condition when warranted by actual or indicated unsafe criticality conditions
- Ensure that engineered and/or administrative criticality safety means of control are in place and functioning properly at the RBOF
- Accept responsibility for the development of a fire safety plan for the RBOF that recognizes, to the extent necessary, both fire safety and criticality safety considerations as specified by applicable DOE and WSRC documents and the Nuclear Criticality Safety Program
- Establish and maintain labeling of fissionable materials in a manner consistent with the specifications of applicable DOE and WSRC documents and the Nuclear Criticality Safety Program and also consistent with the unique features of underwater fuel handling and storage at the RBOF
- Ensure that evacuation routes provide for timely evacuation, that facility changes do not unnecessarily impede or otherwise lengthen evacuation time, and that, to the extent possible, routes do not require personnel to approach potential sites of a criticality accident at the RBOF and neighboring facilities

- Ensure organizational freedom to personnel responsible for training the operations staff, performing health physics duties, or performing quality assurance functions, even if they report to the RBOF Manager
- Provide monitoring and surveillance as necessary to detect and prevent accumulations of fissionable material in unapproved configurations, and take corrective action with input from the CSEG when such accumulations of fissionable material are detected in the RBOF
- Provide criticality safety features and administrative controls as necessary to provide for the criticality safety of the RBOF, including RBOF facility access control
- Approve RBOF response plans, when prepared, for dealing with abnormal situations
- Accept advice from the Facility Operations Safety Committee (FOSC) concerning RBOF criticality safety, and incorporate that advice into decisions that may affect criticality safety
- Be aware of results of reviews and audits provided by the Independent Oversight Review Program and the Assessment Program, and incorporate those findings into RBOF improvements and practices
- Approve revised procedures and temporary procedures in accordance with established administrative procedures
- Through the Emergency Preparedness Manager, ensure that an Emergency Preparedness Program (including appropriate criticality accident responses) is established in accordance with applicable DOE requirements
- Ensure operation of the RBOF in accordance with the approved authorization basis
- Ensure facilitation and control of procedure changes, physical changes in plant configuration, and coordination of the activities of all work groups within the RBOF
- Ensure that the minimum shift crew staffing requirements of the TSR necessary for safe facility operations are adhered to
- Ensure that on-call support personnel are assigned and that technical support personnel are available to provide technical assistance to the RBOF operations staff
- Ensure that RBOF operations are carried out by qualified operators according to written procedures
- Approve any deviations from overtime guidelines necessary to ensure the criticality safety of RBOF facility activities and functions, provided the guideline deviations are documented with a basis in accordance with established procedures
- Ensure that the RBOF first-line supervisors and facility operators are subjected to limitations when being assigned work outside their regular schedules. These limitations are included in Procedure 2.23 of WSRC Procedure Manual 5B (Ref. 38) and Procedure 5.1 of WSRC Procedure Manual 2S (Ref. 39).

The RBOF organization interfaces with various WSRC organizations (not indicated in Figure 6.5-1) in accomplishing the RBOF mission. These organizations include the following:

- Radiological Control Operations (RCO) Section, which provides oversight of the Radiological Protection Program to ensure that the radiation exposure of the facility personnel is maintained As Low As Reasonably Achievable (ALARA)
- Site Safety Review Committee, which meets periodically to assess the adequacy of environment, safety, health, safeguards, security, and quality assurance
- FOSC, which meets periodically to review occurrences and to ensure that significant issues are adequately evaluated

RBOF SHIFT SUPERVISORS

The shift supervisor is responsible for the local command function of the RBOF. During any absence of the shift supervisor from the RBOF, a designated, qualified individual assumes the command function. A turnover of the command function is documented in the operating log. RBOF shift supervisor responsibilities, some as indicated in WSRC-SCD-3 and Chapter 6 of the Generic SAR (Ref. 3, 4), include the following:

- Be knowledgeable in those aspects of criticality safety relevant to operations in the RBOF
- Accept responsibility for the criticality safety of operations in the RBOF
- Ensure that criticality safety training is administered to, and that the appropriate training records are maintained for, RBOF personnel, and require that these personnel have an understanding of procedures and safety considerations necessary to perform their functions without undue risk
- Participate in the On-the-Job Training/Job Performance Measure portion of the fuel handler qualification
- Develop or participate in the development of written procedures applicable to RBOF operations, and be responsible for the maintenance and update of those procedures
- Verify compliance with criticality safety specifications for new or modified equipment prior to its use in the RBOF
- Require conformance with good safety practices, including unambiguous identification of fissile materials and good housekeeping at the RBOF
- Obtain training and assistance, as necessary, from the CSEG

RBOF OPERATORS

RBOF operators have the following responsibilities, as indicated in WSRC-SCD-3 and Chapter 6 of the Generic SAR (Ref. 3, 4):

- Be trained in criticality safety commensurate with their duties, typically consisting of general and RBOF-specific training
- Conduct the On-the-Job Training/Job Performance Measure portion of the fuel handler qualification
- Acquire and maintain fuel handling qualification
- Accept responsibility for criticality safety of their own actions and the operations under their control
- Be familiar with and follow criticality safety procedures as written, and adhere to all nuclear criticality safety steps in operating procedures related to their assignments
- Stop performing a procedure and notify supervision if the procedure does not correspond to the material and conditions upon which it is being performed or if a condition develops that is not addressed by procedures

6.5.1.2 PECD General Manager and/or Chief Engineer Authority and Responsibility

The PECD General Manager and/or Chief Engineer has responsibilities for activities involved in criticality safety in the RBOF, as indicated in WSRC-SCD-3, Chapter 6 of the Generic SAR, and the RBOF TSR (Ref. 3, 4, 18), including the following:

- Ensure that an interface or matrix arrangement exists with SFSD to provide necessary criticality safety expertise to RBOF facility activities and operations. The SRS Criticality Safety Function noted in WSRC-SCD-3 and Chapter 6 of the Generic SAR is provided by the CRT Section of PECD. Within that interface/matrix arrangement, ensure the following:
 - Applicable criticality safety standards and DOE Orders are applied in the design, modification, and operation of RBOF facilities.
 - The WSRC policy for nuclear criticality safety, as stated in MP 4.5 of WSRC-1-01 (Ref. 1) and as implemented in WSRC-SCD-3, is applied at the RBOF.
 - Appropriate training is developed and provided to the RBOF staff in compliance with applicable DOE Orders.
 - Review of each NSDS is provided by CRT personnel not directly or indirectly involved in the generation of the NSDS.

- Delegate PECD criticality safety responsibilities for the RBOF to lower-level managers within the division as necessary, but maintain overall accountability for such responsibilities at the general manager level:
 - Provide qualified PECD personnel to serve as independent criticality expert members of the SFSD Criticality Safety Committee responsible for the RBOF
 - Comply with the nuclear criticality safety design requirements (specified in applicable DOE and WSRC documents) that are applicable to RBOF activities
 - Administer the sitewide NIM Program, including placement and evacuation zone analyses, for the RBOF
 - Provide probabilistic risk assessments of the potential for criticality accidents for SARs, reliability analyses, and other analyses in support of NCSEs related to the RBOF, as requested
 - Ensure that appropriate training relevant to the RBOF is developed and provided to the CSEG staff assigned to the RBOF in compliance with applicable DOE Orders, as needed

CRITICALITY SAFETY FUNCTION PERSONNEL

The SRS Criticality Safety Function noted in WSRC-SCD-3 and in Chapter 6 of the Generic SAR (Ref. 3, 4) is provided by the CRT Section of PECD, which includes a CSEG and a Criticality Technology Group (CTG), and which exists to provide criticality safety engineering services across the site (Section 6.5.1.6 of the Generic SAR addresses responsibilities of the Criticality Safety Function of PECD that are satisfied by the CRT Section of PECD). Several members of the CRT CSEG are matrixed to SFSD with direct responsibility for many aspects of RBOF criticality safety with additional support from the general CSEG organization as needed.

The responsibilities of that matrixed CSEG team include the following:

- Prepare NCSEs and nuclear criticality safety engineering calculations to support the RBOF in accordance with WSRC Procedure Manual E7 (Ref. 40) and WSRC-SCD-3 (Ref. 3)
- Maintain familiarity with criticality safety documentation related to the RBOF
- Maintain familiarity with current developments in nuclear criticality safety standards, guides, and codes, consulting with knowledgeable individuals, as necessary, to obtain required technical information applicable to RBOF operations
- Initiate formal criticality safety documentation of recommended limits for RBOF operations
- Provide assistance in the development, review, and concurrence of procedures and procedural changes affecting criticality safety in the RBOF

- Review and concur with design drawings and drawing changes affecting criticality safety in the RBOF
- Provide technical assistance in the design of equipment and processes affecting criticality safety in the RBOF
- Review and concur with equipment and process changes affecting criticality safety in the RBOF
- Provide first-level independent evaluation of criticality safety control effectiveness for RBOF operations
- Order any RBOF operations that do not have the required level of criticality safety to be brought to a safe condition
- Provide personnel with criticality safety experience to prepare and maintain the site criticality safety manual (WSRC-SCD-3) and to provide interpretation and guidance concerning compliance with the manual for RBOF activities
- Assist in the development of the RBOF Nuclear Criticality Safety Program, including manuals and procedures
- Comply with the site requirements for administrative procedural controls for safety documentation, as specified in WSRC Procedure Manual 11Q (Ref. 35), as applicable to criticality safety documentation for the RBOF
- Maintain familiarity with all RBOF operations for which criticality safety support is provided and for which criticality safety controls are required
- Participate on the SFSD Criticality Safety Committee responsible for the RBOF for the purpose of assessing criticality safety practices, providing in-process reviews and technical support, as necessary
- Review and approve, as well as provide, requested technical information for general and RBOF-specific criticality safety training courses related to RBOF operations
- Review and concur with any fire safety plan for the RBOF
- Examine reports of procedural violations and other deficiencies in RBOF operations for potential improvements of safety practices and procedural requirements, reporting potential improvements to RBOF and SFSD management

The CRT organization also has general responsibility for the above aspects of RBOF criticality safety and may provide personnel to assist the matrixed team. In addition, the CRT organization

has primary responsibility for numerous aspects of criticality safety in the RBOF (derived from the PECD responsibilities), including the following:

- Provide for independent reviews (i.e., technical or design reviews in accordance with WSRC Procedure Manual E7 [Ref. 40]) of NCSEs and nuclear criticality safety

engineering calculations prepared within CRT to support the RBOF. These reviews may be provided by RBOF matrix team personnel.

- Review all NCSEs related to the RBOF, whether prepared by the CSEG or other groups. These reviews may be provided by RBOF matrix team personnel.
- Review and approve all formal documents related to the RBOF that establish CSLs. These reviews may be provided by RBOF matrix team personnel.
- Provide review of each NSDS by CRT personnel who are not directly or indirectly involved in the generation of the NSDS. These reviews may be provided by RBOF matrix team personnel.
- Provide to criticality safety engineers responsible for RBOF criticality safety analyses controlled neutronics codes and data sets, including established bias and bias uncertainties (or data by which bias and bias uncertainties can be constructed), applicable to the RBOF
- Qualify criticality safety engineers responsible for RBOF criticality safety analyses in the use of neutronics codes and neutron/gamma transport codes
- Perform RBOF NIM location analyses and document the results as required by the Nuclear Criticality Safety Program
- Maintain familiarity with current developments in nuclear criticality safety standards, guides, and codes, consulting with knowledgeable individuals, as necessary, to obtain required technical information applicable to RBOF operations
- Comply with the site requirements for administrative procedural controls for safety documentation, as specified in Manual 11Q (Ref. 35), as applicable to criticality safety documentation for the RBOF
- Maintain familiarity with all RBOF operations for which criticality safety support is provided and for which criticality safety controls are required

6.5.1.3 ESH&QA General Authority and Responsibility

The ESH&QA Division General Manager has responsibilities related to criticality safety in the RBOF, as indicated in WSRC-SCD-3 and Chapter 6 of the Generic SAR (Ref. 3, 4), including the following:

- Provide Radiological Control and Health Physics (RC&HP) Department functions for the calibration, use, and control of sources used to perform NIM field response checks, and for the calibration, use, and adequacy of dosimetry and field survey instrumentation used in conjunction with NIM evacuations at the RBOF
- Include criticality safety among the functional areas to be addressed by Facility Evaluation Board reviews and operational readiness reviews at the RBOF

6.5.1.4 Interfaces and Interrelationships with Other Organizations

As indicated in Chapter 6 of the Generic SAR (Ref. 4), the interfaces and interrelationships of the organizations specified above are defined in MP 4.5 of WSRC-1-01 (Ref. 1) and WSRC-SCD-3 (Ref. 3). Some organizational relationships applicable to the RBOF are indicated in the RBOF TSR (Ref. 18). Figure 6.5-1 shows typical interrelationships.

6.5.1.5 Staff Qualifications

The RBOF TSR (Ref. 18) specifies that a Qualification Program be established for RBOF staff in accordance with applicable DOE regulations. The TSR also specifies that a Training and Replacement Program be established and maintained. Training shall include familiarization with relevant industry operational experience.

Staff selection, qualification, and training requirements for SFSD personnel are defined in Procedure RDP 6.01 of WSRC Procedure Manual RD-1 (Ref. 37) (Table 6.5-1). Minimum education requirements, training requirements, and experience requirements are specified for each level of responsibility ranging from operator to department/division manager. RDP 6.01 also provides qualification, certification, and requalification requirements for each level of responsibility. These requirements satisfied by staff personnel are generally in accordance with DOE Order 5480.20 (Ref. 41) special requirements for personnel education, training, and experience, but exceptions may exist. Such individual exceptions are justified by management in writing, and the exception is requested from DOE as provided in RDP 6.01. RDP 6.03 of Manual RD-1 establishes the management and control of training records.

Staff qualifications for CSEG personnel matrixed to the RBOF are determined by PECD management and are appropriate to the needs identified for criticality safety activities associated with the RBOF. Similarly, qualifications for CRT personnel providing criticality support of any kind to the RBOF are determined by PECD management. In general, all criticality specialist personnel assigned to the CRT Section of PECD are qualified to provide criticality safety support to the RBOF. All criticality specialists in CRT have bachelor or advanced degrees in Nuclear Engineering or related disciplines. The CRT technical personnel generally have experience onsite or in the nuclear field before joining the CRT organization. Relatively inexperienced CRT personnel work under the tutelage of a senior experienced specialist familiar with the RBOF or similar facilities. Technical review and management review of all technical work, especially that supporting CSLs, ensure that work by relatively inexperienced CRT personnel meets the high standards of quality and accuracy expected of all CRT activities.

Staff qualifications are also addressed in Chapter 17.

6.5.1.6 Staff Levels

Staffing requirements for the RBOF are established by management based on the scope of operations proposed. The RBOF TSR (Ref. 18) indicates the minimum level of support for various modes of operations in the RBOF. In general, the minimum shift crew will be comprised of an operator, a first-line supervisor, and an RCO technician for the operation mode and a first-line supervisor for the standby mode.

The number of CSEG personnel assigned to directly support RBOF activities is flexible and depends on the scope of criticality safety-related work required at the RBOF and other site priorities for criticality safety-related analyses. PECD (CSEG) and SFSD management cooperate to determine the most efficient use of the CSEG manpower.

Staff levels are also addressed in Chapter 17.

6.5.1.7 Criticality Safety Committees

Several criticality safety committees monitor the RBOF Nuclear Criticality Safety Program and are described in MP 4.5 of WSRC-1-01, WSRC-SCD-3, Chapter 6 of the Generic SAR, and Procedure 1.02 of WSRC Procedure Manual 11Q (Ref. 1, 3, 4, 35). Those committees include the following:

- The Nuclear Criticality Safety Review Committee (NCSRC), sponsored by the Site Chief Engineer, is responsible for fostering and monitoring criticality safety across the site and is established in Charter 6.10 of WSRC-1-01.
- The NIM Committee, reporting to the NCSRC, reviews NIM design changes (in conjunction with the Administration and Infrastructure Division [AID] Chief Engineer, design authority for the NIM unit), troubleshooting, and maintenance and coordinates configuration control of NIM location analysis.
- The SFSD Criticality Safety Committee reports to the NCSRC and is responsible for monitoring fissionable material handling, storage, processing, and disposal activities from a criticality safety perspective for SFSD facilities, including the RBOF. The SFSD Criticality Safety Committee supports criticality safety self-assessments, and performs in-process reviews, of RBOF operations on a periodic basis and reports its findings to the RBOF and SFSD management and the NCSRC. Membership on the SFSD Criticality Safety Committee includes SFSD management and staff personnel (e.g., Operations, Engineering, Quality Assurance), as well as criticality experts from PECD. The charter for the SFSD Criticality Safety Committee is provided in RDP 5.16 of WSRC Procedure Manual RD-1 (Ref. 37).

6.5.1.8 Operations Safety and Independent Reviews and Assessments

The RBOF TSR (Ref. 18) specifies the responsibilities and functions of several review and assessment committees and programs.

The FOSC advises the RBOF Manager on matters affecting operation of the RBOF and associated activities affecting safety. RDP 2.04 of WSRC Procedure Manual RD-1 (Ref. 37) establishes the FOSCs for SFSD, including the RBOF FOSC. Information on FOSC activities is provided to the RBOF Manager and the Independent Oversight Review Program. The FOSC reviews day-to-day activities to ensure that they are conducted in a safe manner. The FOSC uses a cross-disciplinary method review program to review proposed activities to ensure that they are conducted in a safe manner. Elements and details of the FOSC activities are provided in the TSR and RDP 2.04, including USQ evaluations, major facility modifications, TSR changes, and safety-related issues.

The Independent Oversight Review Program provides an integrated review of RBOF activities and programs to provide senior-level management with an assessment of facility operations, effectiveness of facility staff reviews, and recommendations to improve nuclear safety as specified in the TSR.

An Assessment Program is provided by a group independent of the facility as specified in the TSR. This program includes review of the facility Self-Assessment Program. Assessment elements include conformance with the TSR, Quality Program adherence, staff training and qualification, and effectiveness of corrective actions.

6.5.2 INCORPORATING CRITICALITY SAFETY IN PROCEDURES

Operations in the RBOF (handling or storing fissionable materials) are performed in accordance with approved procedures. Such procedures identify the appropriate criticality safety considerations. All expected activities are covered by procedures. Activities involving conditions not covered by a procedure are stopped (in a safe state) until procedures are written and approved to cover the unexpected conditions.

6.5.2.1 Review of Operations

Operations at the RBOF are described in Sections 2.4 and 6.3.1. RBOF operations, in which criticality safety is a consideration, are governed by written procedures. Table 6.4-2 provides a listing of typical operational procedures that have criticality safety significance in the RBOF. Table 6.5-1 provides a typical listing of SFSD administrative procedures that are applicable to the RBOF. Procedures listed in these tables may be revised, deleted, or deactivated, or procedures may be added as needed by operational and administrative improvements and changes. As is evident by the procedure titles and descriptions, these procedures cover startup, operations, and any modifications that may affect criticality safety.

Operations are reviewed at least annually to ensure that procedures are being followed and that process conditions have not been altered. Additional review is provided by the SFSD Criticality Safety Committee and other review/oversight committees described in previous sections.

Deviations from operating procedures and unforeseen alterations in process conditions that affect criticality safety are documented, reported to management, and investigated promptly. Actions are taken to prevent recurrence (see RDP 9.05 of WSRC Procedure Manual RD-1 [Ref. 37]).

The RBOF TSR (Ref. 18) provides administrative requirements for review and approval of procedures.

6.5.2.2 Handling and Storing Fissionable Materials

Fissionable materials are handled and stored in the RBOF basins. Irradiated fuel handling and storage take place under water. Transfers in the RBOF basin consist of moving safe fuel units from one basin to another through canals connecting the basins or moving fuel units between locations within a basin. Once placed in a designated storage position, fuel units are not generally moved until they are due to be shipped to another facility. The movement of fissionable materials in significant quantities (as specified in WSRC-SCD-3 [Ref. 3]) within the boundary of the RBOF complies with the requirements of DOE Order 5480.24 (Ref. 2) and includes an NCSE and the use of approved procedures.

The handling and storage of fissionable materials in the RBOF satisfy the applicable requirements of WSRC-SCD-3:

- Process operations, storage of nonnuclear materials or equipment that is not directly required for fissionable material storage operations, and all other functions not directly a part of normal fissionable material storage operations are excluded from the storage area, if possible.
- Documented inspections and preventive maintenance are performed periodically on fissionable material storage areas to ensure that the components necessary for criticality control are maintained.
- Signs or other devices are used at strategic locations in the RBOF to provide instructions regarding the following:
 - Interpretations of, and required responses to, alarms
 - Evacuation routes
- Fissionable materials are stored in racks (or equivalent fixtures) capable of securely locating stored materials to prevent displacement, to ensure spacing control, and to meet design requirements for criticality safety under normal operational and some credible accident conditions. Floor storage within a storage facility is only permitted where control of location or other safety requirements (equivalent to the safety

provided by storage racks) are inherently provided by the original containers and their restraints.

- Access to the RBOF is controlled by the H-Area access controls.

Transfers between the RBOF and other facilities, either onsite or offsite, are done in approved casks. Casks containing fuel are shipped to the RBOF or are shipped from the RBOF under DOE site transportation guidelines or DOT transportation guidelines. The transferring of fissionable materials conforms with the requirements of WSRC-SCD-3 and WSRC Procedure Manual 19Q (Ref. 34), as indicated in Chapter 6 of the Generic SAR (Ref. 4), and is not addressed further here. Loading of casks in conformance with the RBOF procedures and requirements ensures that the onsite casks satisfy Manual 19Q requirements related to criticality safety.

Section 6.4.3 describes the incorporation of procedural controls into NSDSs and NOPs.

6.5.2.3 Criticality Safety Posting and Labeling

Positive identification of fissionable materials, particularly fissile materials, is essential to criticality safety. Adequate labeling of fissionable materials and clear posting of work and storage areas where fissionable materials are present should be provided to avoid the accumulation of unsafe quantities of such materials.

Posting refers to the placement of signs indicating the presence of fissionable materials, summarizing key criticality safety requirements and limits, designating work and storage areas, or providing instructions or warnings to personnel.

Labeling refers to the placement of clear and positive identifying marks on specific units or batches of fissionable materials (e.g., cans, packages, containers, boxes, reactor fuel assemblies, targets) to prevent them from being mistaken for other materials and to clearly show the type and amount of fissionable materials present.

RBOF POSTINGS

Generic requirements for postings are specified in WSRC-SCD-3 and Chapter 6 of the Generic SAR (Ref. 3, 4). Compliance with some of these posting requirements is not applicable to criticality safety in the RBOF because of the unique features of fuel handling and storage under water.

Postings in the RBOF include the following:

- The presence of fissionable materials in the RBOF facility is posted at the entrances to the RBOF. Posting of the individual basins is not expected to provide any higher measure of criticality safety than just the entrance postings.

- Two specific postings are provided throughout the fuel handling areas, positioned to be readily visible to personnel performing the associated fuel handling operations: (1) "MOVE ONLY ONE FUEL ELEMENT / BUCKET / BUNDLE AT A TIME."; and (2) "FUEL BUCKETS SHALL BE ISOLATED ON FUEL CENTERLINE MARKERS."
- Evacuation routes and instructions are posted to clearly indicate the evacuation routes from the RBOF. Postings are easy to read and located in such a manner that they may be easily seen. The "fissile material symbol," ANSI Standard N12.1-1989 (Ref. 42), is employed on the postings. Postings were developed in accordance with the requirements and guidance in WSRC-SCD-3 and with assistance and concurrence from the CSEG and review and approval by the SFSD Criticality Safety Committee. The postings are of durable construction. Other postings may be provided as needs are identified. Postings of CSLs in the RBOF are not considered practical due to the large number of different fuel types with different grouping and spacing requirements (limits). Procedures are used extensively to handle and position fuel in the storage racks, and all applicable limits are incorporated into those procedures. Postings of limits on RBOF walls, even if in the vicinity of the basins, are judged to be not useful since operators working over the basin would most often be focused on fuel movements under water and the wall postings would not be conspicuous to the operators to serve as reminders. Criticality safety precautions or prohibitions related to firefighting, such as precautions or prohibitions on the use of water, are not posted at the entrance to the RBOF since fuel storage is under water and the addition of water to the arrays is not a factor.

RBOF LABELINGS

Labeling considerations for fissionable materials are specified in WSRC-SCD-3 and Chapter 6 of the Generic SAR (Ref. 3, 4). The appropriate corresponding RBOF fissionable material labeling is reflected by NOP 244-H-4800 (Ref. 43) requirements:

- Labels for reactor elements consist of a unique serial number etched/machined onto each reactor element. Accompanying paperwork/cards specific to each element link the serial number and information about the fuel, such as facility, mass of fissile/fissionable material, enrichment, exposure, and other information pertinent to safe handling, storage, and processing of the element.
- Bundles of fissionable material stored or handled in the RBOF are identified with a unique identification number, which is linked in the inventory records to the material that has been placed in the bundle.
- Discrepant labeling that involves fissionable materials is reported to RBOF management and handled on a case-by-case basis by RBOF management and the CSEG.

6.5.2.4 Evacuation and Treatment Considerations for Criticality Accidents/NIM Alarms

The general evacuation and treatment considerations for criticality accidents/NIM alarms are specified in WSRC-SCD-3 and Chapter 6 of the Generic SAR (Ref. 3, 4). Evacuation and treatment considerations in response to NIM alarms at the RBOF are managed by the H-Area emergency management organization. Visitors to the RBOF are provided a review of the NIM alarm response requirements by reading and signing NOP 244-H-9042 (Ref. 44). Personnel assigned to the RBOF receive such review as part of facility-specific training. Immediate response for visitors and RBOF personnel to NIM alarms is essentially the same. Some specific considerations for the RBOF are as follows:

- Emergency procedures for responding to criticality accidents at the RBOF are contained within the H-Area emergency procedures. Such emergency procedures are approved by area management and the CSEG and are consistent with WSRC-SCD-7 (Ref. 45).
- Personnel assembly stations (i.e., rally points), located outside the areas to be evacuated in the event of a criticality accident, are designated. Figure 6.5-2 shows typical established evacuation routes and rally points for the RBOF. Figure 6.3-1 shows typical evacuation routes within the RBOF/RRF facilities. Process and administrative considerations may require adjustment of evacuation routes and rally points. Such adjustments are reviewed (to ensure personnel safety) and approved by CSEG, Engineering, and RBOF technical and management personnel, as well as the SFSD Criticality Safety Committee, and would be incorporated into RBOF procedures, such as NOP 244-H-9042. The H-Area emergency procedures include provisions for the evacuation of all personnel, care and treatment of injured and exposed personnel, and immediate identification of individuals exposed in a criticality accident. RC&HP procedures provide for personnel dosimetry and instrumentation for determining the radiation level at the rally point and in the evacuated area following a criticality accident.
- Section 6.6 addresses recent evaluations of the need for NIM systems in the RBOF that indicate a criticality alarm system is not effective in the RBOF due to the nature of underwater storage. Conducting operations without the criticality alarm system will require changes to some of the above response and procedural requirements. Responses to a fission product detection system alarm would be similar to the above response and to responses for radiation monitors.

6.5.2.5 Firefighting and Criticality Safety

Water, the firefighting agent used most often, is an efficient moderator and reflector of neutrons. Since fissionable material in the RBOF is handled and stored under water, the use of water to fight fires in the RBOF is not a criticality concern since introduction of water to the basins does not increase moderation or reflection within the basins.

6.5.2.6 Procedure Review, Approval, and Control

The RBOF Manager has established a procedure review, approval, and control program for the RBOF, which includes the following:

- Procedures that involve operational changes related to criticality safety controls or that may require criticality safety controls because of new or planned operational changes are reviewed and approved by CSEG personnel prior to system operation.
- Procedure changes that involve criticality safety controls, or that may require criticality safety controls because of such changes, are reviewed and approved by CSEG personnel prior to system operation.
- Procedure changes that may impact criticality safety are reviewed and approved by CSEG personnel prior to use.
- Engineering drawings should identify equipment and engineered systems important to criticality safety, particularly if such equipment or systems are used as a double contingency defense. Engineering drawings that show equipment and systems important to criticality safety are reviewed by CSEG personnel. Changes to drawings involving equipment important to criticality safety are reviewed by CSEG personnel.
- Supervisors periodically review active procedures. The requirement to periodically review active procedures is itself a procedure. NOP 244-H-1600 (Ref. 46) provides guidance to ensure that TSR requirements are adequately reflected in operational procedures.

Chapters 10 and 12 address the technical content of procedures and the administration of procedure generation, revision, review, and approvals.

6.5.3 CRITICALITY SAFETY TRAINING

MP 1.18 of WSRC-1-01 (Ref. 1) states that it is WSRC's policy to provide its employees with the training and development opportunities necessary for them to complete their work assignments in a safe, effective, and quality manner. Regarding criticality safety training, the training policy supports MP 4.5 of WSRC-1-01 in that all reasonable efforts are taken to reduce or eliminate the potential for, and consequences of, a criticality accident. The training provided to RBOF

personnel complies with the general and specific requirements for criticality safety training established in WSRC-SCD-3 and noted in Chapter 6 of the Generic SAR (Ref. 3, 4).

Administrative requirements are specified in RDP 6.01 of WSRC Procedure Manual RD-1 (Ref. 37) for the selection, criticality safety training, examination, qualification, retraining, reexamination, and requalification of individuals who do the following:

- Work with, handle, or store greater than exempt quantities of fissionable materials, or work with equipment (including construction and maintenance) in which greater than exempt quantities of fissionable materials are processed
- Manage facilities or provide engineering support to facilities in which greater than exempt quantities of fissionable materials are stored, handled, or processed
- Perform special nuclear material accountability functions
- Perform nuclear materials packaging activities
- Perform reviews of criticality safety documentation

These administrative requirements are pertinent to the RBOF. RBOF personnel are trained by the training groups of SFSD.

Chapter 12 also addresses training for RBOF personnel.

6.5.3.1 RBOF Criticality Training Requirements

The overall program for RBOF personnel training has been established by the Training, Procedures, and Support Group of SFSD and is outlined in RBFOPD00X0101 (Ref. 47). That program requires initial and annual training for RBOF personnel, with the level of training dependent upon responsibilities within RBOF operations. Completion of specific training courses is required for operators and supervisors to be considered qualified and/or certified. Specific courses related to criticality safety include the following:

- Nuclear Science & Criticality
- Fuel Handling, Processing, and Storage
- Cask/Fuel Handling

The CSEG personnel matrixed to the RBOF receive PECD- and CSEG-specified training, including appropriate criticality safety topics. All personnel receive basic criticality safety training concerning response to alarm signals as part of annual training.

Visitors to the RBOF are escorted or receive RBOF-specific training for response to a criticality alarm. That training is provided by NOP 244-H-9042 (Ref. 44), which must be signed by the visitor before receiving unescorted access to the RBOF. That training involves the evacuation routes in response to a NIM alarm.

Chapter 12 also addresses training for RBOF personnel.

6.5.4 DETERMINATION OF OPERATIONAL NUCLEAR CRITICALITY LIMITS

6.5.4.1 Nuclear Criticality Safety Evaluation Methodology and Bases

Section 6.5.4.1 of the Generic SAR (Ref. 4) discusses preparation of NCSEs in general, and that information is not repeated here. Specific aspects of the limits generation process for the RBOF are described here, although the general aspects of the RBOF NCSE process is the same as for all other SRS facilities.

An NCSE is an independent, documented analysis that establishes the technical basis for nuclear criticality safety and provides the basis for and recommends subcritical operating limits, criticality safety controls, and engineered criticality safety features. These NCSEs identify the minimum subcritical margin when establishing such limits. NCSEs may also identify specific procedural and hardware limits necessary to implement limits or to document criticality evaluations that do not produce specific limits. An NCSE for the RBOF provides the basis of the development of a new or revised NSDS. WSRC-SCD-3 (Ref. 3) establishes the suggested format and contents for an NCSE. RBOF NCSEs are generally prepared by CSEG personnel matrixed to the RBOF. Sometimes calculations (in accordance with WSRC Procedure Manual E7 [Ref. 40]) that form the basis for an NCSE may be performed by other than CSEG personnel, but those calculations are reviewed and approved (accepted) by RBOF CSEG personnel.

The extensive computer calculations that form the basis of most RBOF NCSEs are performed with established and certified criticality codes (such as those specified in DPSTM-86-700-3 [Ref. 48]) approved by the CRT Section of PECD. CRT regularly reviews the set of computer codes at SRS considered applicable for criticality safety analyses and documents the approved codes (Ref. 49). Computer codes for criticality safety calculations comply with Quality Assurance Procedure (QAP) 20-1 of WSRC Procedure Manual 1Q (Ref. 50) and ECS-SED-940026 (Ref. 51). CRT establishes biases to be used with those codes for various fuel types and conditions and provides the framework within which the applicable biases for those codes can be determined for new fuel in the RBOF (see References 52 through 56). CRT provides training and guidance in the use of those computer codes.

When it is determined that a fuel to be received in the RBOF is not authorized by a current NSDS, the CSEG prepares an NCSE that analyzes proposed handling and storage configurations for the fuel and determines the appropriate spacing and safe number requirements for that fuel in the RBOF needed to develop the NSDS (Figure 6.4-1). Aspects of loading and unloading shipping casks are included in the NCSE, including analyses of dropped assemblies and dropped basket(s), in the vicinity of the cask, as appropriate. An NCSE would also be required to analyze the effect of new equipment associated with fuel handling and storage or of new or revised processes to be established with the fuel or in the vicinity of the fuel. The general requirements for NCSEs are described in WSRC-SCD-3 and Chapter 6 of the Generic SAR (Ref. 3,4). The

description of the fuel and its bounding parameters (e.g., maximum enrichment, maximum initial fissile loading, fissile loading corresponding to maximum reactivity worth [for fuels with plutonium buildup]) are determined from the Appendix A.

Assumptions used in the analyses are documented and justified in the NCSE and generally include the following:

- Since irradiated fuel handling in the RBOF is conducted under water, analyses for the RBOF consider the fuel fully immersed (flooded) in water. If more effective moderator or reflector materials are involved in the situation being evaluated, those materials are modeled.
- When calculating the safe number of fuel elements, assemblies, or bundles in an uncontrolled grouping, the fuels are modeled with water and concrete reflection at the optimum (most reactive) separations.
- During analyses, bounding assumptions are made for moderation, density, enrichment, reflection, and temperature that correspond to the most reactive operating conditions.
- Various degrees of homogenization of the fuel assemblies may be used, related to the degree of homogenization used in establishment of the bias for that fuel type. Homogenization of some fuel types, such as MTR (thin, enriched uranium, U-Al fuel plates), results in conservative calculations of k_{eff} . Homogenization of other fuel types, such as low enriched uranium fuel, may result in unconservative calculations of k_{eff} and accordingly are avoided.
- When analyzing the interactions between rows of fuel, the center-to-center spacing between the rows of fuel is typically modeled as 1/2 inch less than is normally available in the storage racks. This assumption provides allowance for the fact that the spaces between guide plates and between spacers in a fuel row are larger than the bundle cross-sectional dimensions, and thus fuel bundles may not be exactly at the nominal center-to-center spacing of the rows. It also provides some allowance for variations in row I-beam center-to-center spacing.
- When analyzing interactions between fuel along rows and across rows, the effect of fuel assemblies or bundles tilting within the constraints of the hardware is also evaluated. The bundles may be modeled as tilting to touch adjacent fuel and also to within the closest distance of approach to adjacent fuel as limited by the hardware and rack I-beams. If tilting within the hardware results in a calculated k_{eff} exceeding the specified limit for the basin, additional constraints on storage of the fuel must be identified.
- When modeling the fuel in rows, the fuel bundle may be modeled as filled with fuel assemblies, even when the safe number evaluation limits the number of assemblies allowed in a single grouping. This assumption provides a minor conservatism in the calculation of the interaction effects between rows of fuel.

- Fuel contents are always modeled at the most reactive fissile content for the fuel. For most fuels (such as highly enriched uranium fuel), this assumption means that the fuel is modeled at its maximum unirradiated (as-manufactured) fissile loading.
- When fuel may be stored adjacent to several fuel types, the interactions are evaluated for each adjacent fuel type. For adjacent fuel types (e.g., MTR fuel), the interaction may be evaluated for a bounding (most reactive) MTR fuel type.
- The biases, bias uncertainties, margins for area of applicability, and minimum margin of safety used in the evaluation are specified, and the appropriateness of those values is justified.

Having developed the limits and requirements needed to handle and store the proposed fuel in the RBOF, the NCSE addresses the controls needed to ensure compliance with the double contingency principle. Generally, this step would just note that the new fuel requires no controls other than those already provided. If a new process to be initiated in the RBOF were the reason for the NCSE, the controls associated with that process and possible "process upsets" would be detailed to demonstrate the double contingency principle. If acceptance of the fuel at the RBOF or if installation of the new equipment or process would necessitate changes in authorization basis documentation, that would be noted.

The NCSE establishes the limits required in the NSDS (Figure 6.4-1):

- Maximum allowable number of fuel units in worst configuration
- Within-row storage requirements
- Across-row and along-row spacing requirements
- Appropriate storage hardware
- Fuels involved in establishing along-row and across-row storage requirements
- Spacing requirements related to the use of any additional constraints determined necessary (e.g., empty bundles)

Technical review and approval of the NCSE are provided by the CRT Section of PECD. The NSDS is prepared by the CSEG based on the NCSE. Only after the NSDS is approved (see Section 6.5.4.2) is the proposed fuel accepted at the RBOF or the proposed process or equipment operation initiated.

There may be special purpose NCSEs that do not follow the above approach in that they do not involve calculations to determine CSLs. One such NCSE is the Double Contingency Analysis (Ref. 19). The Double Contingency Analysis emphasizes the identification of process upsets and the controls (engineered and procedural) that prevent those upsets from resulting in a criticality incident (see Section 6.4.4). Another example was a specific Tilting Analysis NCSE (Ref. 22), which was written to approve the fuel configurations that existed at the time of the analysis. Specific fuel and hardware configurations that existed at the time of the analysis were acceptable,

taking into account fuel tilting considerations. EPD-CTG-95-0036 (Ref. 57) provides the following interpretations:

- Fuel and hardware configurations that existed at the time of the analysis were acceptable, taking into account fuel tilting considerations.
- Fuel tilting within the hardware type specified within the NSDS was also covered under that analysis.

The second item indicates that, should a fuel be moved from its current row position to another row position with the same hardware type, criticality safety is still maintained, even with fuel tilting, provided the fuels in the adjacent rows (i.e., across-row interactions) are similar to the fuel specified.

6.5.4.2 Nuclear Safety Data Sheets

The NSDS is the base document for nuclear criticality safety control at the RBOF facility. Section 6.4.3.1 describes and discusses elements of an NSDS as does WSRC-SCD-3 (Ref. 3). Figure 6.4-1 shows a typical NSDS. WSRC-TR-95-0302 (Ref. 6) discusses requirements and approvals for NSDSs in addition to the discussions in this section and Section 6.4.3.1.

NSDS COMPLIANCE VERIFICATION

As required in the RBOF TSR (Ref. 18), a program is in place at the RBOF to perform periodic inspections of RBOF storage basins to verify compliance with NSDS requirements (see NOPS 244-H-1602, -1604, and -4823 [Ref. 58, 59, 60]). Specified storage rows are inspected on a periodic basis such that all storage rows are inspected. The periodic inspections include verification that along-row and across-row spacing requirements are maintained and that storage hardware is properly installed/engaged.

REVIEW AND APPROVAL OF NSDSs

Requirements for review and approval of NSDSs are specified in WSRC-TR-95-0302 (Ref. 6). A new or revised NSDS is completed by a member of the CSEG based upon NCSE recommendations. Review and approval of the NSDS are normally provided by a CTG technical reviewer and the Manager, CRT Section. Approvals for the NSDS are then obtained from the Manager, Engineering, and the RBOF Manager. The NSDS is then forwarded to the Regulatory Programs Section of SFSD.

If the NSDS revision is a minor revision, Regulatory Programs distributes the NSDS. If the NSDS revision is a major revision, it is forwarded by Regulatory Programs to DOE-SR for review and concurrence by the Director, Technical Division, or designee. Upon return to Regulatory Programs, the fully approved NSDS is distributed.

6.5.4.3 Interrelationship Between Operational Nuclear Criticality Limits and Technical Safety Requirements

There is no difference between CSLs and operational nuclear criticality limits in the RBOF. The CSLs applicable to the RBOF (safe numbers, in-row spacing requirements, across-row spacing requirements, isolation spacing, identified safe arrays and associated hardware, and adjacent fuels) are determined in NCSEs as specified in WSRC-SCD-3 (Ref. 3) and are then incorporated in NSDSs specific to each fuel type accepted in the RBOF. TSR requirements are directed at implementing the limits specified in the NSDSs. Those NSDS limits are then incorporated into the operational procedures, as appropriate. Thus, the procedural operating limits (NSDS limits) are equivalent to the limits specified in the applicable NCSE. Some NCSE requirements may be incorporated directly in operating procedures.

With the large number of fuel types that are handled and stored in the RBOF, specification of all CSLs in a TSR document would involve frequent and numerically prohibitive changes to the TSR document. The RBOF TSR (Ref. 18) does not specify the CSLs but provides administrative (organizational) requirements that a Fuel Receipt and Shipping Program and a Nuclear Criticality Safety Program be established, implemented, and maintained and that LCOs related to compliance with the CSLs be established. The Nuclear Criticality Safety Program is specified in the TSR to contain the following elements:

- Use of the NSDS as the base document for operational controls
- Fuel Pre-Receipt Program (verify that fuels accepted at the RBOF are within the authorization basis)
- Approved storage locations (analyses have determined handling and storage criteria)

These elements are summarized in Section 6.4.2. An NSDS is required for each fuel type. NSDS usage is discussed in Sections 6.4.3 and 6.5.4.2. As noted in Sections 6.4.3 and 6.5.4.2, each NSDS requires extensive review and approval prior to operation within the requirements of that NSDS.

Surveillances in the TSR verify compliance with the CSLs (Table 6.4-1). Those surveillances require verification of handling and storage activities in compliance with the NSDSs.

6.5.5 CRITICALITY ASSESSMENTS, INSPECTIONS, APPRAISALS, AND REVIEWS

WSRC-SCD-3 and Chapter 6 of the Generic SAR (Ref. 3, 4) discuss and provide an overview of site requirements for assessments, inspections, appraisals, and reviews related to criticality at SRS facilities. Section 6.5.1 summarizes the responsibilities and interfaces of SRS organizations responsible for aspects of criticality safety at the RBOF, including general guidelines and criteria for conducting criticality assessments, inspections, appraisals, and reviews. Specifically, Sections 6.5.1.7 and 6.5.1.8 address the TSR requirements, the SFSD Criticality Safety Committee, the FOOSC, the Independent Oversight Review Program, and the Assessment

Program, noting their involvement in assessments, inspections, appraisals, and reviews. The assessments, inspections, appraisals, and reviews related to criticality at SFSD facilities (including the RBOF) are addressed in various procedures of WSRC Procedure Manual RD-1 (Ref. 37) (Table 6.5-1).

Chapter 17 of the Generic SAR (Ref. 61) and Chapter 17 of this SAR provide additional information concerning safety reviews and performance assessments, including those in the area of criticality safety.

6.5.5.1 Assessments and Inspections

Section 6.5.1.8 discusses the RBOF TSR (Ref. 18) requirements for the FOSC, the Independent Oversight Review Program, and the Assessment Program. The TSR addresses the elements of each of those programs. These programs provide for either RBOF self-assessments or appraisal of the self-assessments, with each program requiring a different level of independence from the operations staff and management. The criticality safety self-assessments are conducted at least annually and are defined as reviews of operations important to criticality safety or of progress on issues and findings from previous self-assessments, DOE appraisals or reviews, and RBOF walkthroughs. The SFSD Criticality Safety Committee may assist in performing criticality safety self-assessments.

Criteria for performing self-assessments are provided in WSRC-SCD-4 (Ref. 62). The Site Assessment Program is described in MP 1.22 of WSRC-1-01 (Ref. 1). Instructions for performing self-assessments are provided in Management Requirement and Procedure (MRP) 5.15 and MRP 5.20 of WSRC Procedure Manual 1B (Ref. 63).

Inspections are conducted periodically to verify compliance with the NSDS handling and storage requirements. These inspections are intended for compliance with TSR surveillance requirements (Table 6.4-1) and are proceduralized (Table 6.4-2).

6.5.5.2 Appraisals

The ESH&QA Facility Evaluation Boards perform independent oversight of criticality safety self-assessments performed by the RBOF. (See MRP 5.15 and MRP 5.20 of WSRC Procedure Manual 1B [Ref. 63] and Section 6.5.1.8.)

DOE-SR and DOE Headquarters may conduct periodic criticality safety appraisals of facilities at SRS, including the RBOF.

6.5.5.3 Reviews

The NCSRC and SFSD Criticality Safety Committee conduct criticality safety reviews, as appropriate, of revisions in operations, new or modified systems/equipment, limit violations, identified or perceived hazardous conditions, etc.

Reports are prepared presenting the results of a criticality safety review and are distributed to RBOF management.

MRP 5.09 of WSRC Procedure Manual 1B (Ref. 63) requires a triennial management review of the site Nuclear Criticality Safety Program.

6.5.5.4 Recordkeeping

RBOF recordkeeping requirements and retention policies are addressed in the RBOF TSR (Ref. 18). SFSD has established a formal documented recordkeeping system (RDPs 15.02, 15.03, and 15.04 [Ref. 37] as indicated in Table 6.5-1) for the control and retention of criticality safety documents and records. RBOF procedure tracking and maintenance are established in accordance with WSRC Procedure Manual 2S (Ref. 39). The CRT Section of PECD uses the document and record management systems established by the WSRC Sitewide Records Inventory and Disposition Schedule in conjunction with the SRS Quality Assurance Program.

The criticality safety document and record management systems consider the following items:

- NCSEs and NSDSs are considered criticality safety documents and records.
- Operating procedures and TSRs that contain information important to criticality safety are considered criticality safety documents and records.
- The CRT Section has established a system for uniquely numbering and archiving NCSEs.
- An RBOF procedure (NOP 244-H-1607 [Ref. 33]) serves as a comparison document between operating procedures and the TSR for comparing criticality safety requirements, limits, steps, and/or conclusions. Each NSDS directly references the NCSE upon which it is based.
- SFSD has established systems to link functional tests and new (or revised) as-built drawing dimensions back to the documents that serve as the basis.
- SFSD has established a system (see procedures of WSRC Procedure Manual RD-1 [Ref. 37] in Table 6.5-1) for tracking the status of the following:
 - Assessment deficiencies
 - Corrective actions
 - Facility Evaluation Board and operational readiness review findings

- DOE appraisals
- Other assessments, inspections, appraisals, and reviews related to criticality safety
- CSEG review and approval of NSDSs and incorporation of NSDS limit values into operating procedures ensure that all necessary approvals are obtained prior to authorizing the use of processes, operations, procedures, or equipment whose safe performance is dependent (in whole or in part) upon such criticality safety documents.

6.5.6 CRITICALITY EMERGENCIES AND LIMIT VIOLATIONS

6.5.6.1 Criticality Infraction Reporting and Followup

The RBOF TSR (Ref. 18) addresses reporting of conditions and violations adversely involving criticality safety in accordance with DOE regulations. The TSR identifies conditions that constitute TSR violations and conditions not considered by TSR requirements that may require reporting to DOE regulatory authorities.

DOE is notified of operating violations involving criticality-related TSRs in accordance with WSRC Procedure Manual 9B (Ref. 64) and site reporting requirements, as appropriate. Reporting also ensures that other operations may benefit from lessons learned.

RBOF operating procedures generally govern immediate actions to be taken, in the event of an unanticipated situation, to place the operation into as stable and safe a condition as possible until a criticality safety engineer/specialist can conduct an evaluation. Such actions generally involve stopping the movement of fissionable material and isolating the affected material. If a criticality control step in an operating procedure or a criticality (NSDS) limit is violated, activities controlled by the procedure are discontinued immediately, unless it is unsafe to do so, and supervision is notified. Identification of a limit violation generally necessitates a stoppage of all fuel handling activities until the violating condition has been investigated and restored to compliance with applicable limits.

Administrative procedures in WSRC Procedure Manual RD-1 (Ref. 37) (Table 6.5-1) provide reporting and documentation guidelines as to required reporting periods and authorities. Those administrative procedures also establish the review and administrative processes necessary for recovery from the abnormal condition. WSRC-SCD-3 (Ref. 3), Chapter 6 of the Generic SAR (Ref. 4), and Section 6.5.1 address the responsibilities of SFSD (RBOF) and PECD CRT (CSEG) personnel and management in responding to procedural and TSR infractions and violations, including participation in followup activities such as critiques and lessons learned documentation.

The CSEG reviews and approves all recovery procedures for recovery from a limit violation.

RBOF emergency procedures clearly specify reporting responsibilities and duties of the Facility Emergency Coordinator and the Area Emergency Coordinator in the event of a criticality accident or a NIM alarm of unknown origin. Chapter 15 of the Generic SAR (Ref. 65) and Chapter 15 of this SAR contain information concerning emergency procedures.

6.5.6.2 Recovery from a Criticality Infraction

The RBOF TSR (Ref. 18) requires that an Emergency Response Program be established. As noted above, some operating procedures provide immediate response to some abnormal conditions. Other emergency procedures provide immediate response to emergency conditions (e.g., NIM alarms).

The TSR also requires that a Response Plan Program be established at the RBOF. A response plan involves one or both of two functions: (1) to restore the RBOF to compliance with LCOs, and (2) to ascertain what additional actions are needed to ensure that the RBOF is in compliance with TSR requirements and Nuclear Criticality Safety Program requirements. Response plans may be prepared in anticipation of postulated events (or classes of events).

Procedures are developed, as appropriate, to govern actions to be taken in the event of an unanticipated criticality infraction. The procedures' objectives are to ensure that the operation is placed into a stable/safe condition until a criticality safety engineer can conduct an evaluation. As noted above, most RBOF procedures have initial response actions addressed within the procedure.

Recovery from violations of CSLs is accomplished in a manner to ensure that the remaining safety margin is acceptable or is not further reduced if already unacceptable. If the situation permits, safe shutdown and recovery from a CSL violation are conducted under an approved procedure consistent with existing requirements to conduct all operations in accordance with written procedures.

Procedures for recovery from a CSL violation will assign specific responsibilities for actions to be taken during the recovery, including the role of the Facility Emergency Coordinator, the Area Emergency Coordinator, the recovery team, and the Technical Support Center.

6.5.6.3 Classification, Investigation, and Reporting

Deviations from procedures and unforeseen alterations in process conditions that affect criticality safety are reported to management and investigated promptly.

Violations of CSLs, criticality safety steps in procedures, criticality safety procedures, TSRs, or equipment failures are classified as either an "off-normal occurrence," an "unusual occurrence," or an "event" and are handled according to WSRC Procedure Manual 9B (Ref. 64).

An unplanned critical excursion is classified as an "event" and is handled according to WSRC-SCD-3 (Ref. 3).

6.5.6.4 Corrective Action

Following recovery from a CSL violation, permanent corrective actions are developed and implemented to reduce the probability of a violation recurrence (e.g., prepare better procedures, install more reliable equipment, improve training, install additional passive-engineered controls).

Following recovery from a criticality accident or from a near criticality accident (i.e., loss of control to the extent that no known reliable mechanisms to prevent a nuclear criticality are functional), corrective actions are developed and implemented to effectively preclude the recurrence of the event.

Lessons learned are incorporated into the facility's Training Program and nuclear safety analysis, as appropriate, for either of these events. WSRC Procedure Manual RD-1 (Ref. 37) procedures (Table 6.5-1) establish aspects of a Lessons Learned Program for SFSD.

Chapter 11 of the Generic SAR (Ref. 66) provides additional information concerning corrective actions.

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6.6 CRITICALITY INSTRUMENTATION

6.6.1 CRITICALITY DETECTION AND ALARM SYSTEMS REQUIREMENTS

Criticality instrumentation consists of devices to detect criticality incidents, to provide alarms in the event of an inadvertent criticality, and to provide measures of the extent of exposures and releases due to the incident. Criticality instrumentation does not prevent any criticality accidents, but provides means to alert personnel to the occurrence (and to initiate evacuation to minimize exposure) and provides means to characterize the extent of the criticality accident.

WSRC-SCD-3 (Ref. 3) requires consideration of installation of these devices in compliance with DOE Order 5480.24 (Ref. 2) and ANSI/ANS-8.3-1986 (Ref. 12) in all facilities that handle fissionable material. WSRC-SCD-3 provides detailed guidance to ensure compliance with the DOE Order 5480.24 requirements. The SRS criticality alarm system is designated as the NIM system.

DOE Order 5480.24 requires compliance with portions of ANSI/ANS-8.3-1986 and requires a criticality alarm system to be installed when the mass of fissionable material may exceed established subcritical mass limits (see paragraph 4.2.1 of ANSI/ANS-8.3-1986) and when the probability of criticality is greater than $1\text{E-}06/\text{yr}$. That alarm system must cover occupied areas in which the expected dose exceeds 12 rad in free air. The alarm system includes a criticality accident detection device and a personnel evacuation alarm. An alarm system is not required in a storage basin if sufficient shielding to protect personnel (dose from initial burst <12 rad) exists, but a fission product detection system should be provided in adjacent occupied areas.

An SRS criticality alarm system is installed at the RBOF. That criticality alarm system uses NIMs. The overall NIM system is the responsibility of Central Services Works Engineering of AID. Requirements for the SRS NIM system (capabilities and coverage) are established in WSRC-SCD-3.

Exemptions to the criticality alarm system requirements are discussed in the requirements documents [including WSRC-SCD-3 (Ref. 3)] and in Chapter 6 of the Generic SAR (Ref. 4), but have not been addressed for the RBOF until recently.

6.6.2 DESCRIPTION OF CRITICALITY DETECTION AND ALARM SYSTEMS IN RBOF

A general description of the SRS NIM system is provided in Section 6.6.1 of the Generic SAR (Ref. 4). Documents referenced in the Generic SAR discuss the following and provide guidance for establishing coverage zones (region in which detector responds to minimum accident of concern):

- Characteristics of the NIM detector and associated electronics
- Required sensitivity (response) of the detector

- Alarm logic of the system
- Failure modes of the system
- Testing and response experience of the NIM detector unit

The RBOF NIM system uses the NIM pair system described in the Generic SAR, in which either detector unit may initiate an alarm. No further description of the NIM system is provided here.

A NIM system is installed in the RBOF even though the RBOF is a storage basin in which fuel handling and storage takes place under water and, in general, the water above the fuel provides significant shielding to reduce radiation exposure to personnel in the vicinity. That NIM system was installed following the general guidance that a NIM pair should be installed within 100 feet of any location that may contain sufficient fissionable material to support a criticality, with a reduction in that distance if there is intervening shielding between the potential accident location and the detector pair. The distances were measured horizontally and the NIM pairs were generally positioned so that structural walls and such shielding were not between potential accident locations and the NIM pairs. The shielding due to the depth of the basin water was not taken into account.

Figure 6.3-1 shows the relative positions of the NIM units and alarm bells in the RBOF. Those units are positioned to be within 100 feet (horizontal distance) of any location containing fissile material in the basins. A NIM pair is positioned at the west wall of the #1 storage basin, about 30 feet north of the south wall of the #1 storage basin, and about 6 to 7 feet above the surface of the water in the basin. A second NIM pair is similarly positioned just north of the repackaging basin while a third NIM pair is similarly positioned just north of the inspection basin. Alarm bells are provided on each NIM unit, and a remote alarm system provides additional alarm bells (nine) throughout Building 244-H. A NIM alarm will be indicated in the H-Area Control Room.

6.6.3 NIM SYSTEM DESIGN REQUIREMENTS

The NIM Committee (see Section 6.5.1.7) is chaired by the AID Chief Engineer and is responsible for SRS standardization of NIMs, specification of maintenance routines and schedules, and development of procedures by which design modifications for the instruments may be authorized. The NIM Committee assigns responsibility for maintenance, installation, and troubleshooting to appropriate organizations at the facilities. Other NIM Committee responsibilities are contained in its charter and are described in WSRC-SCD-3 (Ref. 3). Detailed NIM design requirements, including the selection of equipment functions and sensitivities, are described in WSRC-SCD-3 and Chapter 6 of the Generic SAR (Ref. 4).

6.6.4 NIM TESTING

WSRC-SCD-3 (Ref. 3) specifies requirements, including frequencies of performance, in the following areas associated with NIM testing:

- Initial testing and inspections of NIM systems
- Testing and inspections of NIM systems following significant modification to, or repair of, a NIM system
- Response checks of NIM systems
- Recalibration of NIM systems
- Testing of NIM alarm systems
- Testing of NIM bell circuits
- Field audibility tests of NIM signals
- Checks of NIM alarm decibel levels
- Procedures for NIM testing
- Corrective actions when tests reveal inadequate NIM system performance
- Records of NIM system tests, recalibrations, and corrective actions

NOPs 244-H-1616, -9043, and -9044 (Ref. 67, 68, 69) are in place to perform that testing and maintenance.

6.6.5 NIM LOCATION, EVACUATION BOUNDARY, AND SOURCE TERM ANALYSIS

General responsibilities for NIM location and evacuation analyses are addressed in WSRC-SCD-3 and Chapter 6 of the Generic SAR (Ref. 3, 4) and are not repeated here. The CSEG has responsibility for those analyses.

Figure 6.3-1 indicates the current location of NIM pairs and alarm bells in the RBOF. The RBOF NIM pair locations are based on the generic NIM response analysis. The NIM system was installed in the RBOF even though an exemption for the RBOF could have been based on the shielding provided to personnel by the basin water. At that time, the water shielding over the fuel was considered to be about 10 feet deep. That depth of water would reduce worker radiation doses from criticality incidents to insignificant levels. ANSI/ANS-8.3-1986 (Ref. 12) indicates that a criticality alarm system is not required for fuel storage basins when the water depth is such to sufficiently limit the radiation dose to personnel. At the time of installation, reduction in NIM response for some small criticality accidents due to the water shielding was not directly assessed.

6.6.5.1 Dose to RBOF Workers from Criticality Incidents

Current analyses, N-NCS-H-00010 and N-NCS-H-00011 (Ref. 70, 71), have evaluated the direct dose levels potentially received by workers in the RBOF due to a criticality incident occurring in a storage basin. Examination of the storage configurations in the RBOF determined that, although most fuel in the basin has about 10 feet of water shielding between the upper portion of the fuel and the upper surface of the basin water, some fuel bundles may have fuel within 5.33 feet of the basin water surface (Ref. 72), resulting in potentially insufficient shielding to reduce dose to RBOF workers to less than 12 rad. Detailed calculations using the Monte Carlo N-Particle Transport Computer Code have modeled several postulated accident conditions within 4.5 feet of the basin water surface and at several other depths in the water. The analyses (Ref. 70, 71) considered the fission yields expected for process handling criticality accidents ($5\text{E}+17$ fissions) and NPH-initiated criticality accidents ($5\text{E}+18$ fissions).

When the top of the interacting fuel is shielded only by 4.5 feet of water, the estimated dose to workers on the traveling bridge, directly over the incident location, exceeded the 12-rad limit specified to define the need for an alarm system in accordance with ANSI/ANS-8.3-1986 (Ref. 12). As the water depth over interacting fuel increases, the dose is dramatically reduced. With a water shield of at least 7 feet, the dose received by a worker on the bridge crane is less than 12 rad, even if the NPH-induced fission yield of $5\text{E}+18$ fissions is assumed. Figure 6.6-1, derived from N-NCS-H-00010 and N-NCS-H-00011, shows the dose to personnel on the traveling bridge, as a function of water depth over fuel, for a process-related criticality and for an NPH-related criticality. Although the EBS racks were not installed at the time of this analysis, the water coverage over the EBS racks is approximately 2 feet more than the non-EBS racks, and the above analysis assumptions are maintained for fuel stored in the EBS racks.

N-NCS-H-00011 indicates that only a process-related criticality incident is of concern for fuel with a water depth less than 7 feet since an NPH-related criticality will occur only below 7 feet. Thus, credible criticality incidents in the RBOF basin will not result in a dose to personnel working over the basin (on the traveling bridge) exceeding 12 rad. N-NCS-H-00011 also evaluated the potential dose to personnel positioned about the RBOF other than on the traveling bridge and concluded that the water depth and the concrete wall and earth shielding about the basin preclude personnel doses of 12 rad or more. Since a 12-rad zone cannot be defined for the RBOF, use of a criticality alarm system in the RBOF is not needed. This is justifiable under the DOE Order 5480.24 (Ref. 2) requirements outlined in Section 6.6.1.

The potential dose to RBOF workers from a criticality in the basin is less than 12 rad. Thus, exception can be taken to the need for a NIM system in the RBOF. Installation of a fission product detection system is recommended by DOE Order 5480.24. Engineering is evaluating the option to deactivate (remove) the RBOF NIM system and to employ a fission product detection system.

6.6.6 NIM SYSTEM OPERATION

NIM system operation is described in WSRC-SCD-3 and Section 6.6.6 of the Generic SAR (Ref. 3, 4) and is not repeated here. An LCO in the RBOF TSR (Ref. 18) is established for NIM operability. RBOF operating procedures incorporate these requirements for NIM operability during fuel handling. If one or more NIM units monitoring a potential accident location either are found to be out of service as a result of a malfunction, are going to be unavailable because of authorized preventive maintenance, or need to be temporarily bypassed or taken out of service, all operations involving fissionable material at the location are suspended or brought to a controlled and orderly conclusion until full NIM coverage is restored.

6.6.7 NUCLEAR ACCIDENT DOSIMETRY

Fixed nuclear accident dosimetry units are installed in the RBOF consistent with DOE Order 5480.11 (Ref. 73). The locations of the fixed Criticality Neutron Dosimeters (CNDs) are specified in WSRC-IM-92-118 (Ref. 74). That manual is prepared and updated appropriately by RCO, and copies are maintained in each RCO office. Table 6.6-1 lists typical locations for CNDs assigned to the RBOF.

Personnel nuclear accident dosimeters meeting the requirements of DOE Order 5480.11 (Ref. 73) are worn by all personnel entering the basin area of the RBOF (Ref. 3).

Chapter 7 of the Generic SAR (Ref. 75) discusses the role of accident dosimetry in determining the impacts of a criticality incident.

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6.8 TABLES

Table 6.3-1 RBOF Typical Storage Bundles

Bundle Type	Material	Length (inches)	Dimension (inches)	Ref.
Square Tube Dissolver Bundle	Aluminum	168	3.54 x 3.54	76
General Purpose Tube Bundle	Aluminum	168	OD = 5	77
EBS Bundle	Aluminum	144	OD = 5.05	78
SRP ED Scrap Can	Stainless Steel	120	OD = 4.125	79
TRR Canister	Aluminum	145 or 131	OD = 5, 4, or 3	80
EBR-II Container	Aluminum	157	OD = 4.737	81
SRE Canister	Aluminum	110	OD = 3.5	82
HTRE Capsule	Aluminum	36	OD = 4	83

OD - outside diameter

^aNot an all-inclusive listing of storage bundles or cans. (Refer to design drawings for accurate measurements.)

Table 6.3-2 RBOF Hardware Description (Ref. 10)

Hardware	N-S Width (in)	E-W Width (in) ^a	Spaces	# of tiers	V-Trough	Top Angle	Drawings (S5-2-xxxx)	Comments
Adjustable	*	1.5	125	3	Yes	Yes	4300, 4301, 4302, 4303	* = Variable
EBWR 3.75	4	4.125	47	2	No	Yes	W235130	
Hallam 5 1/4	5.25	5.25	36	1	Yes	No	3856, 3857	Install in 12-inch rows
Hallam 5 1/4	5.25	5.25	36	1	Yes	No	4856, 4857	Install in 9-inch rows
HFIR	18.5	38	5	4	No	No	3503, 3504	
HWCTR 2.3	2.625	2.875	67	1	No	Yes	3864, W235084	
HWCTR 3.47	3.75	3.875	42	1	No	Yes	W234936	
Idaho 1-7/8	1.875	2	96	3	No	Yes	3416, 3419, 3421	
Idaho 3-3/8	3.375	3.5	55	3	No	Yes	3416, 3418, 3421	
Idaho 3-7/8	3.875	4	48	3	No	Yes	3416, 3417, 3421	
Idaho 5-1/4	5.25	6	32	2	No	Yes	3416, 3420, 3421	
Idaho 5-1/4	5.25	6	32	3	No	Yes	3533	
Idaho 5-7/8	5.875	6	32	2	No	Yes	3416, 3421, 3422	
MTR	3.5	3.625	53	3	No	Yes	2484, 2485, 2486	
MTR-9T	3.875	3.875	49	1	Yes	No	2750, 2751	
MTR-12T	3.875	4	48	1	Yes	No	2753, 2754	
Piqua	5.625	6	32	1	No	No	2923, 2924	Grating trough width is 4.75 inches
PM-3A	11	4	48	2	No	Yes	3411, 3412, 3413	Grating trough width is 6.5 inches
PM-3AB	4	4	48	2	No	Yes	3413, 3414, 3415	
Vallecitos	3.25	3.5	55	2	No	Yes	2537, 2538, 2539	
Wide - 9T	5.25	5.25	36	2	Yes	No	4858, 5006	Grating trough width is 5.375 inches
Wide - 12T	8.25	4.125	47	2	No	No	5858, 5007	
Bucket*	20	31.5	2	7	No	No	4854, 4855	* = Row 0
Bucket *	20	31.5	2	7	No	No	2756, 2757, 2759	* = Rows 1, 2, 3, 4
Bundle *	3.875	4	12	1	Yes	No	2810, 2811	* = Mk Bun-R hardware type
Bundle *	4	4	12	1	No	No	2810, 3526	* = Mark 12-4 hardware type
Bundle *	4.5	4.5	11	1	No	No	2810, 3526	* = Mark 12-4-1/2 hardware type
Sm. Test Tube*	7	10.75	3	2	No	No	4308, 4310	* = Mark TT 11 inches
Bg. Test Tube							4309	
#2 Basin	6.5	1-51/64	60	2	No	No	6298, 6299	Grating trough width variable
#2 Basin*	4.75	1-51/64	60	2	No	No	6298, 6299	* = with auxiliary guide plate
General							W233806	Basin Area Plan
							W234274	Storage Racks Plan & Sections
							W234280	Details Sheet 1
							W234936	Details Sheet 2
							W235130	Details Sheet 3
							W2350846	Details Sheet 4
							W235120	Details Sheet 5
							3862, 3863	Rack Anchor & ID Markers
							4160, 4161	#2 Basin Arrangement
							4853	Mockup Rack
Row Locks							2813	12-inch Row
							3858	Guide Plate
							5236	Adjustable Hardware

^aDistance = centerline to centerline of spacer

Table 6.3-3 Example Fuel Matrix for Rows A-S and AA-WW

Type	NSDS	Class	Fissile / Assy (g)	Active length (in)	Can Type	Area (in ²)	wt% ²³⁵ U	Form	Cmp'nd	Rows	Bundles in Row	Hardware	N _{safe}
ORR	122	MTR	307	23.63	STDB	9.9	93.5	Plates	U3O8-Al Cermet	A	17	MTR-9T	6
ORR- 20%	217	MTR	347	24	STDB	~9.9	20.6		U-Al-Si	J	12	MTR-9T	6
RINC	164	MTR	130	24	STDB	9.0	93.5	Plates	UAl	A H	6 7	MTR-9T	10
ATSR	221	MTR	160	23.6	STDB	9.7	93.5	Plates	UAl	A	4	MTR-9T	9
UVA	101	MTR	201	24	NS	9.5	93.5	Plates	UAl	G M	4 4	MTR-9T	7
MIT	59	MTR	514	22.38	STDB	6.5	93.5	Plates	UAl cermet	H	19	MTR-9T	3
HOR	182	MTR	194	24	STDB	9.5	93.5	Plates	UAl	I	9	MTR-9T	4
UM	34	MTR	171	24	STDB	9.8	93.5	Plates	UAl	J	8	MTR-9T	7
Nereide	204	MTR	182	23.5	STDB	9.3	20.5	Plates	U-Al-Si	J	8	MTR-9T	9
SFF	69	MTR	200	23.5	STDB	9.0	93.5	Plates	UAl or U ₃ O ₈ -Al cermet	M	34	MTR-9T	8
ANLJ	222	TUBE	141	23.8	STDB	7.1	93.5	Tubes	UAl	M	4	MTR-9T	9
Astra	195	MTR	311	23.6	STDB	9.6	93.5	Plates	UAl	I	5	MTR-9TW	6
R-2	42	MTR	257	23.5	STDB	9.0	93.5	Plates	UAl	I	11	MTR-9TW	7
DR-3	185	MTR	190	23.5	GP	9.8	93.5	Tubes	UAl	R	5	MTR-9TW	7
JMTR	189	MTR	284	29.9	STDB	9	93.5	Plates	UAl	KK	15	MTR-12T	7
SRE	173	UMO	4500	110	Cans 3.5" OD	9.6	10	Slugs Cyl	UMO UC UO ₂ UO ₂ - THO ₂	A WW	5 32	MTR-9T PM-3AB	2 cans
Dresden	105	Oxide	3224	116.7	Encap Can	11.8 58.4	2.6	Rods	UO ₂ UO ₂ - THO ₂	UU -	18	PM-3A	1 container
Dresden	104	Oxide	1317	116.7	SS sleeves & cans	18	93.5 3 % UO ₂	Rods	UO ₂ - THO ₂	UU QQ	12	PM-3A Hallam 5 1/4	100 rods
CVTR	113 (13 rods / bundle)	Oxide	1027	95.4	STDB	6.7	4.7	SS or Zirc Rods	UO ₂	QQ	3	Hallam 5 1/4	3

Table 6.3-3 Example Fuel Matrix for Rows A-S and AA-WW (continued)

Type	NSDS	Class	Fissile / Assy (g)	Active length (in)	Can Type	Area (in ²)	wt% ²³⁵ U	Form	Cmp'nd	Rows	Bundles in Row	Hardware	N _{safe}
TRR	215	U Rods	2699 553 Pu	120	TRR Can	19.6	.7 .15 % Pu	Rods	NAT U	VV QQ PP OO NN EE CC	23 1 27 30 14 19 29	MTR-9T Hallam 5 1/4 Hallam 5 1/4 Idaho 5 1/4 Hallam 5 1/4 Idaho 5 1/4 Idaho 5 1/4	76 rods
ERR	190	UO ₂ rods	1395 / assy	60	GP	19.6	93.2 (5.53)	Rods SS clad	UO ₂ - THO ₂	TT RR	25 13	MTR-9TW Hallam 5 1/4	128 rods 3 bundles
EBR II	166	MOX	NS	11	Special	9.6	93.5	Rods SS clad	UO ₂ - PuO ₂	DD-3	1	Idaho 5 1/4	76 rods
	209 209	U-Pu metal	1700	156	Special	18.7	0.2	Slugs	U-Pu metal	BB AA	28 31	Idaho 5 1/4 Idaho 5 1/4	4 canisters
B&W	140	Oxide	12 U5 45 Pu	34.5	Special	4.9		Rods, ss clad	UO ₂ - PuO ₂	LL	1	Idaho 5 7/8	1 can
HWCTR Driver Test Mk-6	18 (1-16, 20,21, & 27)	SRS Zr-clad U-Zr fuel	varies, max = 1080	~120	None	NS	NS	Multi Rods Tubes Slugs	Multi	B	44 9 1	HWCTR 2.3	Varies, min = 6
SFO	198	UO ₂ in cans	200	11.1	STDB	4.9	93.5	Cans	UO ₂	C D E G K L N	35 32 33 26 33 34 33	MTR-9T	6 cans 3 bundles
Saxton Cans	150 (Can 28, Can29)	Oxide Scrap fuel	695 Pu 2150 U	50	SS can	9.6	7 12.5	Rods	PuO ₂ in UO ₂ UO ₂	DD-3	2	Idaho 5 1/4	1 can 51 rods
Saxton Bundles	157 172	Oxide See NSDS for details	105 g/rod 32 g Pu/rod	35 - 36.6	SRP ED BT	NS	44	Rods	PuO ₂ in UO ₂ UO ₂	MM	11	Idaho 5 1/4	1 bundle 51 rods 1 bundle 86 rods
HTRE	85	UO ₂ cermet, ceramic SCRAP	1600	36		12.6	93.0	Capsul e	UO ₂ cermet, ceramic	DD-3	13	Idaho 5 1/4	1 capsule
GCRE	92	UO ₂ pellets	968	22	3.5 OD can	9.62	93	Bundle of Pins	UO ₂ / BeO ₂	DD-3	6	Idaho 1 7/8	3 cans or bundles 5 assy
	64	Oxide	812	22	SS Can	1.64	93			DD-2,1	4, 62		
ML-1	64	Like GCRE	810	22	SS Can	1.64	93	Rods	UO ₂ / BeO ₂	DD-2 DD-1	39 29	Idaho 1 7/8	5 assy
MURR	119	MTR	785	24	GP	10	93.5	Curved Plates	UAl	F O Q	23 24 22	MTR-9TW	2 assy 3 bundles

Table 6.3-3 Example Fuel Matrix for Rows A-S and AA-WW (continued)

Type	NSDS	Class	Fissile / Assy (g)	Active length (in)	Can Type	Area (in ²)	wt% ²³⁵ U	Form	Cmp'nd	Rows	Bundles in Row	Hardware	N _{safe}
EBWR	199	Oxide Plates	50.5 / plate	34	Any	24	93.5	FF Plate	UO ₂ SS cermet	RR	4	Hallam 5 1/4	NL
	179	Oxide Rods	817 Pu	48	Al Cans, 3.75 x 3.75	14	.236 U	Rod ZR clad	PuO ₂ - UO ₂	LL-2 LL-1	5 20	Idaho 5 7/8 4 1/2 Adj	4 cans 300 rods
	124	Oxide Rods	917 1613	48 48	As Assy or can	12.7	93.2 6	Rods, Zr clad	UO ₂	JJ-2, 1 II-2 HH-2, 1	30, 31 31 25, 31	4 1/2 Adj 4 1/4 Adj 4 1/4 Adj	4 assy 81 rods
	125	U-Zr-Nb Plates	810	58	4.25 x 4.25 cans	14	1.44	Plates, Zr clad	U-Zr-Nb	II-2, 1 HH-2 GG-2, 1	31, 39 25 40, 41	4 1/4 Adj 4 1/4 Adj 4 1/4 Adj	11 assy
GTRR	165	MTR	188.5	23.5	STDB	8.22	93.5	Plates, Al Clad	UAl	G	5	MTR-9T	8
OSU	220	MTR	185	24	STDB	9.0	93.5	Plates, Al Clad	UAl	H	4	MTR-9T	6
GRR	218	MTR	194.4	23.6	STDB	9.75	93.5	Plates, Al Clad	UAl	H I	6 5	MTR-9T	7
SAPHIR	181	MTR	290	25.8	STDB	9.49	93.5	Plates, Al Clad	UAl	I	6	MTR-9T	6

NS = None Specified, NL = No Limit

The above list is not all inclusive and is provided for information.

Table 6.4-1 Examples of TSR Surveillance Requirements Related to Criticality Safety

Surveillance Requirement
Verify RBOF basin water level is higher than -18 inches with respect to the RBOF floor grade
Verify three RBOF NIM pairs are in service
Perform Instrument Loop Check on each of the RBOF NIM units
Perform Instrument Loop Test on each of the RBOF NIM units
Perform Instrument Loop Calibration on each of the RBOF NIM units
Perform Decibel Check on the RBOF NIM bells
Verify remote 244-H General Alarm is in service
Perform Instrument Loop Test on 244-H General Alarm
Verify remote NIM Alarm in 221-H Control Room is in service
Perform Instrument Loop Check on remote NIM Alarm in 221-H Control Room
Verify fuel in the # 1 or # 2 storage basin that has been moved is stored per NSDS requirements or is isolated
Verify fuel in other than the #1 or #2 storage basin is stored as isolated critically safe quantities or is in a cask
Verify fuel stored in the #1 or #2 storage basin is stored per NSDS requirements or is isolated
Verify cask baskets containing greater than the safe number of fuel pieces that may be stored in a Safe Fuel Unit are not removed from the cask until an engineering evaluation has determined removal of the baskets is safe
Verify supporting grating for upper tier fuel is installed at NSDS required spacing above the lower tier before fuel is moved into the upper tier (Upper tier fuel may be moved if NSDS determined operation without vertical spacing is safe)
Visually verify that significant quantities of material that are better reflectors than water that are not part of a fuel unit are maintained 12 inches from fuel unless the material is needed for operations and an evaluation has determined the presence of the material is safe
Verify an NSDS applicable to the fuel type being handled has been prepared, documented, and approved
Verify NSDS compliance for all fuel types
Verify 100-ton crane is positioned north of the #2 storage basin
Perform Instrument Loop Test on 100-ton crane stop switch
Verify 100-ton crane stop bar actuates crane stop switch
Verify fuel identification matches the identification provided by the shipping facility
Verify shipping requirements are met for 70-ton cask shipments
Verify cadmium plates are installed in SRS 70-ton cask when required per the applicable NSDS
Verify cadmium plates in 70-ton cask reduce neutron transmission to 50% in neutron transmission test

Table 6.4-2 RBOF NOPs Related to Criticality Safety (Example Listing from RBOF Facility Procedure Index)

Procedure #	Procedure Title
244-H-1013	RBOF Routine Water Samples
244-H-1102	Building 244-H and 245-H Daily Shutdown and Startup
244-H-1109	Transferring Loads Between 100-Ton Crane & Monorail Hoist or Chainhoist
244-H-1131	100-Ton Crane Operation
244-H-1132	Traveling Bridge Operation
244-H-1133	Monorail Operation
244-H-1210	Operating of Underwater Video and Camera Equipment
244-H-1237	RBOF Interim Storage Program
244-H-1241	RBOF Operator Round Sheets
244-H-1302	Two-Person Rule at 244-H
244-H-1600	User's Guide for RBOF Technical Safety Requirements
244-H-1602	Periodic Inspection Checklist for Surveillance Inspections/Verifications
244-H-1604	RBOF Facility Surveillance Inspection Checklist - All Modes
244-H-1607	RBOF TSR/PR Surveillance Documentation Cross Reference
244-H-1615	Criticality Rebaseline NCR Disposition
244-H-1616	244-H Nuclear Incident Monitor Instrument Loop Test
244-H-3000	Handling Spent Fuel Shipments Between Offsite Sources and SRS
244-H-3005	Multiple procedures for Receipt and Shipment of Casks
3012	- PEGASE
3014	- BMI-1
3015	- SRP 70-Ton Cask
3053	- TN7-2
3054	- GNS II
3719	- NLI-1/2
	- NAC LWT
244-H-3025	Cask Receiving and Shipping Report
244-H-3033	Cask Receiving, Shipping and Inspection Record Intraplant Transport Only
244-H-3502	Changing Inserts in 70-Ton Cask
244-H-3504	Loading the 70-Ton Cask
244-H-4000	Certification of Fuel Receipt Nuclear Safety
244-H-4001	Storing Fuel in Either Storage Basin Directly From a Cask

Table 6.4-2 RBOF NOPs Related to Criticality Safety (Example Listing from RBOF Facility Procedure Index) (continued)

Procedure #	Procedure Title
244-H-4002	Removing Fuel From Storage Racks and Transferring to a Working Basin
244-H-4003	Transfer of Fuel From One Working Basin to Another
244-H-4004	Relocating Fuel in Storage Rows Within Storage Basin
244-H-4005	Storing Fuel Bundle Assemblies in Either Storage Basin
244-H-4020	Bucket Storage Activities to a Working Basin
244-H-4200	Nuclear Material Control & Accountability at RBOF
244-H-4212	RBOF Physical Inventory and Random Sampling
244-H-4800	Cask Unloading and Fuel Packaging
244-H-4818	Guidelines for 244-H Fuel Handling Activities
244-H-4820	Removing Fuel From a Cask for Transfer to a Working Basin
244-H-4822	Encapsulation Ruptured Taiwan Research Reactor (TRR) Fuel Bundle
244-H-4823	Verification of Approved Fuel Storage Configurations
244-H-5304	Cropping and Bundling of Fuel Assemblies
244-H-7001	Operating the Bandsaw
244-H-7021	Inspection of Square Tubes and General Purpose Tubes Bundling Tubes
244-H-9042	RBOF Visitor Criticality Review
244-H-9043	NIM Monitor and Alarm Test
244-H-9044	RBOF/RRF Nuclear Incident Monitor Alarm and Response and Testing of the Remote Alarm System
244-H-9593	Alarm Response Procedure for the RBOF Facility
244-H-9595	RBOF Response to Health Protection Monitors

Table 6.4-3 Hazard Analysis Events Related to Criticality - RBOF

Event	Event Description	Cause
34	Floating fuel container resulting in criticality	Storage violation, equipment malfunction, defective container
35	Fuel storage infraction results in criticality	Procedure violation or worker error
36	Deformation of Fuel Storage Rack results in criticality	Roof and/or crane falls into Storage Basin due to external load dropped from construction crane
37a	Deformation of Fuel Storage Rack results in criticality	Mechanical failure causing crane or crane load to fall into basin
37b	Deformation of Fuel Storage Rack results in criticality	Worker error causes hoist hook to snag rack, dropped fuel
38	Critical configuration in fuel bundle causes criticality	Worker error in assembling fuel bundle
39	Configuration of fissile debris on basin floor causes criticality	Fuel ruptured, damaged, or corroded
40	Fuel spills from fuel basket during fuel movement causing criticality	Basket dropped or ruptured due to basket lid off during moving, worker error, equipment malfunction
40a	Fuel spills from shipping cask during fuel loading/unloading operations resulting in criticality on basin floor	Equipment malfunction or operator error causes cask lid to fall onto cask
41	More than one basket out of cask at same time resulting in criticality	Procedure violation
42a	Fast leak in 70-ton cask adds positive reactivity resulting in criticality	Natural phenomena, vehicular impact, or equipment mishandling
43	Low water level in basin allows decay heat to deform fuel to unsafe geometry causing criticality	Leakage of water from basin to subsoil due to earthquake or rack failure
44	Disassembled pieces are arranged in an unsafe geometry on basin floor resulting in criticality	Worker error during disassembly, procedure violation
63	Earthquake causing damage to the basin walls, or storage racks, roof collapse, or crane derailed resulting in a criticality	Natural phenomena
63a	Event 63 coupled with basin draindown	Natural phenomena
64	Tornado causes criticality due to roof collapse	Natural phenomena
65	Tornado or high winds impel missile into storage basins causing a criticality by dislodging stored fuel into an unsafe geometry	Natural phenomena

Table 6.4-4 Double Contingency Analysis - RBOF Criticality Scenarios (Ref. 19)

Criticality Scenario	Bounding Assumption/CSL Violated
Fuel received is more reactive than expected. Handling per specified CSLs may result in criticality	Bounding Assumptions on mass/assembly, fuel enrichment, fuel type
Cask loading error by vendor	SARP required spacing or presence of neutron absorber
Cask dropped during placement in cask basin resulting in fuel spill	CSL, number in group, violated and/or Bounding Assumption on absorption
Load dropped from crane during inadvertent crane travel over storage basin. Impact with rail stops cause loss of load. Operator error in crane operation	Analyzed safe storage configuration disrupted.
During cask basket handling, fuel drops back into cask or on floor	CSL on safe number. CSL on spacing.
Multiple errors in fuel unit handling resulting in multiple fuel units dropping back into cask.	CSL on spacing.
Error in cask lid removal resulting in lid drop into loaded cask	CSL on geometry.
Overhead crane falls on open cask during unloading	CSL on spacing and/or geometry.
Error in loading the unloading bucket exceeds safe number of fuel units	CSL on mass (safe number).
Error in specification of safe number of fuel units	CSL on mass.
Error in loading the transfer bucket exceeds safe number of fuel units	CSL on mass.
Cask basket removed from cask in presence of other fuel	CSL on spacing.
Transfer bucket loaded in the presence of other fuel	CSL on spacing.
Criticality due to cask spacing violation	CSL on spacing.
Incorrect number of fuel units specified for inclusion in empty bundle and placed in bundle	CSL on mass.
Incorrect number of fuel units added to empty bundle	CSL on mass.
Incorrect number of fuel units specified for inclusion in partial bundle and placed in partial bundle	CSL on mass.
Incorrect number of fuel units added to partial bundle	CSL on bundle mass.

Table 6.4-4 Double Contingency Analysis - RBOF Criticality Scenarios (Ref. 19) (continued)

Criticality Scenario	Bounding Assumption/CSL Violated
Incorrect partial bundle selected, based on erroneous content data, and filled with excess assemblies	CSL on mass.
Incorrect partial bundle for assembly addition retrieved due to incorrect specification and filled with excess assemblies	CSL on mass.
Accumulation of incorrectly cropped ends in scrap bucket or fissile fines on floor in vicinity of saw	CSL on mass or geometry.
Error in specification of storage position results in bundle/assembly stored in more reactive location than analyzed	CSL on spacing. Bounding Assumption on adjacent fuels
Placement of fuel in incorrect storage row results in bundle/assembly stored in more reactive location than analyzed	CSL on spacing. Bounding Assumption on adjacent fuels
Failure to maintain required empty positions within storage row	CSL on spacing.
Failure to maintain HFIR components only on post in HFIR storage structure	CSL on spacing violated
Placement of fuel in proximity to incorrectly documented fuel	CSL on spacing.
Error in calculational data to support storage position selection (tilt) and NCSEs erroneously approve unsafe fuel arrays in storage racks	CSL on spacing.
Failure to correctly position required rack hardware results in inadequate spacing control between rows of assemblies or individual assemblies	CSL on spacing.
Inadvertent storage in non-approved location	CSL on spacing.
Failure to maintain required spacing between buckets during temporary storage in the working basins	CSL on spacing.
Hoist hook operated improperly resulting in fuel rack displacement	CSL on spacing.
Hoist hook improperly positioned during bridge operation resulting in rack displacement	CSL on spacing.
Failure to respond to out of range basin chemistry reading	CSL on geometry or CSL on spacing.
Failure to detect out of range basin chemistry reading	CSL on geometry or CSL on spacing.
Fuel moved incorrectly in canal resulting in spacing violation	CSL on spacing.
Fuel moved incorrectly in disassembly basin resulting in spacing violation	CSL on spacing.
Fuel moved incorrectly in repackaging basin resulting in spacing violation	CSL on spacing.
Fuel moved incorrectly in cask basin resulting in spacing violation	CSL on spacing.

Table 6.4-4 Double Contingency Analysis - RBOF Criticality Scenarios (Ref. 19) (continued)

Criticality Scenario	Bounding Assumption/CSL Violated
Fuel moved incorrectly in storage basin aisle resulting in spacing violation	CSL on spacing.
Erroneous fuel storage at storage row end resulting in spacing violation during transport	CSL on spacing.
Personnel error in fuel unit handling results in fuel drop in storage matrix.	CSL on spacing.
Mechanical failure during fuel movement results in drop in storage matrix.	CSL on spacing.
Error in loading the transfer bucket exceeds safe number of fuel units	CSL on mass.
Material in basin/canal results in enhanced reflection or moderation	CSL on moderation, reflection.
Error in specification of the number of fuel units to be placed in a cask.	CSL on mass.
Excess fuel units placed into the cask.	CSL on mass.
Error in the location of a fuel unit in a cask	CSL on spacing.
Error in selection of fuel for cask loading	CSL on mass.
Error in loading cask resulting in fuel unit dropping into the cask.	Bounding Assumption on cask internals (spacings, poisons, mass).
Filled cask dropped during removal from cask basin resulting in fuel spill	CSL, number in group, or Bounding Assumption on absorption
Error in cask lid placement resulting in lid drop into loaded cask	CSL on geometry.
Overhead crane falls on open cask during loading	CSL on spacing and/or geometry.
Error in placement of absorbing material within cask	CSL on absorption.
Error in verification of absorbing material effectiveness	CSL on absorption.
Water leaks from cask after removal from basin resulting in change in moderation/ reflective properties (wet shipped casks)	CSL on internal cask moderation, reflection.
Earthquake deformation of storage rack resulting in unsafe fuel spacing.	CSL on spacing
Earthquake induced damage results in unsafe fuel spacing	CSL on spacing
High wind induced structural damage results in unsafe fuel spacing	CSL on spacing
High wind (tornado) induced missile results in unsafe fuel spacing	CSL on spacing

Table 6.5-1 RD-1 Procedures With Potential Criticality Safety Significance (Ref. 37)

Procedure ID	Procedure Purpose
RDP 1.01	Defines divisional organization and general responsibilities at the division and department levels
RDP 1.06	Provides the SFSD process for preparing action plans to resolve concerns formally transmitted from DOE that require an action plan to document the division corrective actions due to complexity of the corrective actions, multiple corrective actions, or DOE request
RDP 1.07	Provides for a Master Tracking System, a computerized database to track divisional workloads, commitments, issues, work items, action items, etc., to ensure identification and visibility of work, and to provide communication to enhance effective work
RDP 1.09	Establishes the Reactor Independent Safety Review Committee, which provides senior management oversight of industrial, operational, and nuclear safety. Procedure is committee charter. The FOSC within SFSD reports to this committee.
RDP 1.12	Establishes responsibilities and requirements for division response to DOE Stop Work Order
RDP 2.04	Establishes the FOSC within the division, delineating functions and responsibilities, membership, member qualification and training, meetings, review and evaluation, and tracking and documentation
RDP 2.05	Establishes the division stance on fitness for duty, establishing guidelines for response to situations involving fitness for duty
RDP 2.06	Establishes requirements for individuals performing procedure walkdown validation
RDP 2.08	Provides instructions for isolation and restoration of equipment and systems to ensure safety of personnel and equipment during maintenance, inspections, and test activities
RDP 2.11	Establishes responsibilities and requirements for preparation, review, approval, revision, and control of interpretations made to SFSD facilities Technical Specifications, Plant Operation Specifications, Technical Standards, and Technical Safety Requirements
RDP 2.14	Provides guidance for testing activities per QAP 11-1 and WSRC 5E manual, including test results to determine acceptance, rejection, performance, or disposition of a structure, system, or component; surveillance testing; functional testing; acceptance testing; post-modification testing; and special testing
RDP 2.23	Provides the Process Requirements for RBOF, delineating operation and surveillance requirements in addition to the operation and surveillance requirements in the TSR
RDP 5.02	Establishes the Request for Reactor Engineering Assistance form and associated administrative controls as the formal mechanism to request technical, design, and engineering evaluations from Engineering Department and is intended to be used to obtain technical information and evaluations of potential problems when sufficient information or expertise is not available
RDP 5.16	Establishes the Criticality Safety Committee within the division, delineating functions and responsibilities, membership, member qualification and training, meetings, review and evaluation, and tracking and documentation

Table 6.5-1 RD-1 Procedures With Potential Criticality Safety Significance (Ref. 37)
(continued)

Procedure ID	Procedure Purpose
RDP 6.01	Establishes selection, training, and qualification requirements for personnel involved in monitoring, maintenance, and technical support of SRS production reactors while in cold standby status. Specifically applicable to 100-Area personnel. May be revised to include RBOF personnel
RDP 6.03	Establishes requirements and responsibilities for management and control of training records within SFSD
RDP 6.04	Specifies the components of the Employee Indoctrination Program to be satisfied by SFSD personnel
RDP 6.05	Establishes requirements and responsibilities for selection and designation of subject matter experts for SFSD facilities
RDP 7.01	Establishes responsibilities and requirements for preparation, review, revision, approval, cancellation, and control of SFSD administrative procedures
RDP 7.02	Establishes responsibilities and requirements for preparation, review, revision, cancellation, approval, deactivation, immediate change requests, and control of SFSD technical procedures. Manual 2S, Procedure 1.1, "Procedure Administration," is specified to be used in RBOF
RDP 8.01	Provided a system for identifying and resolving nonconforming items with SFSD. Procedure deleted and QAP 15.1 of 1Q Procedure Manual used instead.
RDP 9.05	Defines the process for tracking, maintaining status, and closing corrective actions originating from Occurrence reports required by DOE Order 5000.3B
RDP 10.04	Establishes administrative controls for processing correspondence received by SFSD from DOE or the Defense Nuclear Facilities Safety Board and establishes controls for continuing operational commitments
RDP 12.03	Defined responsibilities and described the USQ Determination process as implemented in SFSD to satisfy DOE Order 5480.21. Procedure deleted and Procedure 1.05 of 11Q Procedure Manual used instead.
RDP 15.02	Establishes responsibilities and requirements for the control of documents generated, received, and maintained by the SFSD personnel that are designated as controlled documents
RDP 15.03	Defines and establishes the requirements of the Records Management Program for SFSD
RDP 15.04	Defines and establishes the requirements for correspondence control in SFSD
RDP 17.04	Defines the SFSD program for controlling and maintaining individual and collective personnel radiation exposure, releases, and radiological hazards to ALARA levels

Table 6.6-1 Locations of Fixed Criticality Neutron Dosimeters for the RBOF (Ref. 74)

CND Location Number	Location
RBOF 1	Building 244-H basin exit
RBOF 2	Traveling Bridge
RBOF 3	West of Cask Wash Pit
RBOF 4	Truckwell Door 109
RBOF 5	Repackaging Basin
RBOF 6	Control Room East Wall
RBOF 7	North of Building 244-H (Outside)
RBOF 8	West Wall (Outside Building 245-H)
RBOF 9	South of Building 244-H
RBOF 10	Building 244-H Entrance Hard Structure
RBOF 11	East of Building 244-H (Outside)
RBOF 12	North Fence (Outside)
RBOF 13	South Fence (Outside)

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6.9 FIGURES

Figure 6.3-1, RBOF/RRF (244-H/245-H) Building Diagram and Evacuation Routes (showing NIM and Alarm Bell Locations), has been removed and may be found in WSRC-TR-95-0302, Nuclear Criticality Safety Manual -Spent Fuel Storage Division Facilities.

Figure 6.3-1 RBOF/RRF (244-H/245-H) Building Diagram and Evacuation Routes

Figure 6.3-2, Schematic of RBOF Storage Basins, has been removed and may be found in WSRC-TR-95-0302, Nuclear Criticality Safety Manual -Spent Fuel Storage Division Facilities.

Figure 6.3-3 RBOF Storage Hardware Schematic (Not to Scale)

NUCLEAR SAFETY DATA SHEET 189	DATE 5-12-97	WSRC-TR-95-0302 Appendix Data Sheet: 189 Rev.: 4	Page 1 of 1								
Reactor: Japan Materials Test Reactor (JMTR)		Applicability (Check All That Apply) ☒ RBOF ☒ L ☐ K	Latest Rev. Status ☐ Minor ☒ Major								
Fuel Type: ☒ High Enriched U-Al ☐ Other Fuel Description: JMTR fuel assemblies have a square cross section, aluminum side plates, and flat, aluminum clad U-Al fuel plates. Before irradiation, the U in the fuel is no more than 93.5 wt% U-235. There are two different assembly types - the standard fuel assembly and the fuel follower assembly. The standard fuel assembly measures about 7.704 cm on a side, has an active fuel length of about 75.9 cm, holds 19 fuel plates, and contains a maximum of 283.88 g of U-235. The fuel follower assembly measures about 6.364 cm on a side, has an active fuel length of about 75.0 cm, holds 16 fuel plates, and contains a maximum of 198.56 g of U-235.											
Nuclear Safety Requirements: Prepared By: <u>Joe Bryce</u> Date: <u>5-12-97</u> 1. Safe Number of Fuel Pieces in Water: <u>7 (seven) assemblies</u> Concrete Reflection Considered? ☒ Yes ☐ No (Fuel for which concrete reflection has not been considered may only be handled one piece at a time.) 2. When outside the storage racks, a safe number of this fuel assembly type must be separated from all other fuel by 12 inches of water or 24 inches of concrete. 3. Allowed Storage Locations: ☒ RBOF Basin #1 or Basin #2 (Rows YA - YK) Allowed Storage Bundle: ☐ GP Tube ☒ Square Tube ☐ Disassembly Bundling Tube a) Maximum number of fuel pieces per bundle: <u>5 (five)</u> (Bundle capacity may reduce this number) b) Hardware: <u>MTR-9T & MTR-12T</u> c) Safe Fuel Group in Storage Row: <u>No Limit</u> d) May be stored in any <u>9 or 12</u> inch row for which interaction requirements are satisfied, or Only Row: <u>N/A</u> is acceptable e) Number of empty spaces between Safe Fuel Groups: <u>None</u> Empty Spaces must be filled with a <u>N/A</u> f) Number of empty spaces between this fuel and other approved fuels in same row: <u>1 for 9" rows; N/A for 12" rows</u> Empty Spaces must be filled with a <u>square tube dissolver bundle</u> g) Adjacent row approved fuel types must be stored in <u>9 (nine)</u> inch (or greater) rows. h) Other approved fuel types (Indicate NSDS number, "High Enriched U-Al types", etc.) Same Row: <u>For 9" rows - High Enriched U-Al types & 188 (SFO); For 12" rows - None</u> Adjacent Row: <u>For 9" rows - High Enriched U-Al types & 188 (SFO); For 12" rows - (98, 124, 179) EBWR & (140) B&W</u> ☒ RBOF Basin #2 Expanded Storage Racks Allowed Storage Bundle: ☐ GP Tube ☐ Square Tube ☒ Disassembly Bundling Tube a) Maximum number of fuel pieces per bundle: <u>4 (four)</u> (Bundle capacity may reduce this number) b) Storage in any rack location? ☒ Yes ☐ No (See Page 2) (Fuel bundles must be verified to be closed) ☒ L-Basin Expanded Storage Racks Allowed Storage Bundle: ☐ GP Tube ☐ Square Tube ☒ Disassembly Bundling Tube a) Maximum number of fuel pieces per bundle: <u>4 (four)</u> (Bundle capacity may reduce this number) b) Storage in any rack location? ☒ Yes ☐ No (See Page 2) (Fuel bundles must be verified to be closed) 4. Approved for Shipment in SRS 70-Ton Cask? ☐ Yes ☒ No Allowed Configurations: ☐ Cadmium Plates ☐ MTR Fuel Insert ☐ Other (See Page 2) Cask positions approved for fuel: a) Cadmium Plates b) MTR Fuel Insert Max. number of bundles per approved cask position: a) Cadmium Plates b) MTR Fuel Insert Blocking devices must be installed in the following cadmium plate cask positions: 5. Special Requirements? ☐ Yes (See Page 2) ☒ No											
References: 1) Appendix A Agreement #s 32 & 33 for JMTR Assemblies and Appendix A Agreement DOESRAAF-96-013, Rev. 1, May 8, 1997 (also for JMTR assemblies). 2) Calculations by T. A. Reilly documented in July 1978 on pages 35-48 of WSRC-NB-95-134, Unclassified Notebook CB-4. 3) Raymond L. Reed, "NCSE: Reactivity Effects of Tilting Fuel Assemblies and Bundles in RBOF", EPD-CTG-950028, 3-29-95. 4) Marc A. Rosser & Joe R. Bryce, "NCSE: Storage and Handling Limits for Japan Materials Testing Reactor (JMTR) Fuel in RBOF and L Basin", N-NCS-G-00003, 5-12-97.											
Approvals: <table border="0"> <tr> <td><u>Raymond L. Reed</u> 5/12/97 Technical Reviewer Date</td> <td><u>M. R. Skues</u> 5/19/97 Manager Engineering Date</td> <td><u>William B. Springer</u> 5/17/97 Manager Basin Operations Date</td> <td><u>[Signature]</u> 5/27/97 DOE, Technical Date</td> </tr> <tr> <td><u>[Signature]</u> 5-12-97 Manager, CRT Date</td> <td></td> <td></td> <td></td> </tr> </table>				<u>Raymond L. Reed</u> 5/12/97 Technical Reviewer Date	<u>M. R. Skues</u> 5/19/97 Manager Engineering Date	<u>William B. Springer</u> 5/17/97 Manager Basin Operations Date	<u>[Signature]</u> 5/27/97 DOE, Technical Date	<u>[Signature]</u> 5-12-97 Manager, CRT Date			
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<u>[Signature]</u> 5-12-97 Manager, CRT Date											

Figure 6.4-1 Example of a Nuclear Safety Data Sheet

Figure 6.5-1 Functional Organization Chart for RBOF Organization

Figure 6.5-2 RBOF (244-H) Typical Evacuation Routes

Figure 6.6-1 Dose (Rad) from RBOF Criticality versus Water Depth

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 7

RADIOLOGICAL PROTECTION PROGRAM

AUGUST 1997

UNCLASSIFIED
DOES NOT CONTAIN
UNCLASSIFIED CONTROLLED
NUCLEAR INFORMATION

ADC &
Reviewing
Official: _____

J.N. DEWES, Manager, Regulatory Programs

Date: _____

J. Dewes
9/23/97

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 7

RADIOLOGICAL PROTECTION PROGRAM

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UNCLASSIFIED DOES NOT CONTAIN UNCLASSIFIED CONTROLLED NUCLEAR INFORMATION	
ADC & Reviewing Official:	<u>J. Dewes</u>
J.N. DEWES, Manager, Regulatory Programs	
Date:	<u>9/23/97</u>

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

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ACRONYMNS

ALARA	As Low As Reasonably Achievable
DOE	Department of Energy
FARMS	Facility Annual Review Monitoring System
RBOF	Receiving Basin for Offsite Fuels
RC&HP	Radiological Control and Health Physics
RDP	Reactor Division Procedure
RRF	Resin Regeneration Facility
SAR	Safety Analysis Report
SFSD	Spent Fuel Storage Division
SRS	Savannah River Site
WSRC	Westinghouse Savannah River Company

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7.0 RADIOLOGICAL PROTECTION PROGRAM

7.1 INTRODUCTION

7.1.1 OBJECTIVE

The purpose of this chapter is to provide information on radiological protection that satisfies the requirements of Department of Energy (DOE) Order 5480.23, paragraph 8.b.(3)(i), as amplified in Attachment 1, paragraphs 4.f.(3)(d)9 and 4.f.(3)(d)11a, of the Order (Topic 9) (Ref. 1). This chapter also includes information, as applicable, that partially satisfies the requirements of paragraphs 8.b.(3)(b), (f), and (u) of the Order, which are addressed in each chapter of the Safety Analysis Report (SAR) as necessary to provide explicit correlation to topics (Ref. 2).

7.1.2 SCOPE

This chapter provides information on the Radiological Protection Program for the worker and the public as it applies to the Receiving Basin for Offsite Fuels (RBOF) and the Resin Regeneration Facility (RRF), which are part of the Spent Fuel Storage Division (SFSD), at the Savannah River Site (SRS). The scope of this chapter includes the following:

- Description of the radiological protection organization
- Descriptions of the policies and programs for reducing radiation exposures to values that are As Low As Reasonably Achievable (ALARA)
- Description of radiation exposure control, including administrative limits, radiological practices, dosimetry, and respiratory protection
- Identification of the Radiological Monitoring Program to protect workers, the public, and the environment
- Discussion of the radiological protection instrumentation
- Description of the program for maintaining records of radiation sources, releases, and occupational exposures

Chapter 7 of the Generic SAR (Ref. 3) describes the sitewide Radiological Protection Program at SRS. This chapter presents facility-specific information, referencing the Generic SAR, as well as other chapters, as appropriate.

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7.2 REQUIREMENTS

The RBOF and RRF, as part of SRS, are required to comply with 10CFR835 (Ref. 15) and a number of DOE Orders that contain the basic requirements for radiological protection. Section 7.2 of the Generic SAR (Ref. 3) discusses these requirements.

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7.3 RADIOLOGICAL PROTECTION PROGRAM AND ORGANIZATION

The Radiological Control and Health Physics (RC&HP) Department is the SRS consulting authority for radiological protection of site personnel and the public. RC&HP establishes the Radiological Protection Program for the RBOF and RRF. Section 7.3 of the Generic SAR (Ref. 3) describes the RC&HP organization, program objectives, and experience/qualification requirements.

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7.4 ALARA POLICY AND PROGRAM

7.4.1 ALARA POLICY CONSIDERATIONS

The RBOF and RRF follow the SRS ALARA policy as described in Section 7.4 of the Generic SAR (Ref. 3). The SFSD, which includes the RBOF and RRF, has an SFSD ALARA Program that is consistent with the site policy, which is implemented by Procedure RDP 17.04 of WSRC Procedure Manual RD-1 (Ref. 4).

7.4.2 ASSOCIATED RADIOLOGICAL GOALS

The SFSD Vice President and Manager is responsible for guiding the SFSD in defining and setting ALARA goals. The Facility Manager is responsible for developing ALARA goals (Ref. 4).

Chapter 7 of the Generic SAR (Ref. 3) details SRS plans, which are applicable to the RBOF and RRF, on the establishment of radiation exposure goals and documentation of goal performance.

Key radiological performance indicators are reviewed periodically to determine if goals have been met. If actual exposures deviate by more than 25% from the goals, a further review is conducted and presented to the SFSD ALARA Committee (Ref. 4).

7.4.3 PERSONAL PROTECTIVE PRACTICES AND TECHNIQUES

The RBOF and RRF use the personal protective practices and techniques discussed in Section 7.4 of the Generic SAR (Ref. 3).

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7.5 RADIOLOGICAL PROTECTION TRAINING

The RBOF and RRF follow the site program for training and frequency of training. Section 7.5 of the Generic SAR (Ref. 3) discusses the following types of radiological protection training:

- General Employee Radiological Training
- Radiological Worker Training
- Specialized Radiological Worker Training
- Radiological Control Technician Training
- Subcontracted Radiological Control Technician Training
- Supervisor and Manager Training

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7.6 RADIATION EXPOSURE CONTROL

7.6.1 ADMINISTRATIVE LIMITS

The RBOF and RRF follow the SRS administrative control levels as described in Westinghouse Savannah River Company (WSRC) Procedure Manuals 5Q and 5Q1.1 (Ref. 5, 6). The SRS Administrative Limits Program is detailed in Section 7.6 of the Generic SAR (Ref. 3), which includes the following topics:

- Administrative control provisions
- Administrative control levels
- Radiological worker dose limits
- Embryo/fetus dose limits
- Special control levels
- Planned special exposures
- Emergency and accidental exposures

7.6.2 RADIOLOGICAL PRACTICES

The RBOF and RRF follow the SRS radiological practices as described in WSRC Procedure Manuals 5Q and 5Q1.1 (Ref. 5, 6). Section 7.6 of the Generic SAR (Ref. 3) details the SRS Radiological Practices Program.

7.6.3 DOSIMETRY

The RBOF and RRF follow the SRS Dosimetry Program as described in WSRC Procedure Manual 5Q (Ref. 5). Section 7.6 of the Generic SAR (Ref. 3) discusses the SRS Dosimetry Program, including the types and uses of dosimetry.

7.6.4 RESPIRATORY PROTECTION

The Manager of the Industrial Safety and Hygiene Section has overall responsibility and authority for the Respiratory Protection Program at SRS. Chapter 8 provides additional information on the Respiratory Protection Program. WSRC Procedure Manual 4Q1.6 (Ref. 7) provides details of the Respiratory Protection Program.

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7.7 RADIOLOGICAL MONITORING

As facilities that process/handle particulate or gaseous byproducts or enriched material, the RBOF and RRF have a Facility Annual Review Monitoring System (FARMS) as described in Procedure 458 of WSRC Procedure Manual 5Q1.2 (Ref. 8). FARMS is a joint venture between RBOF and RRF management and RC&HP personnel. FARMS considers the criteria for the protection of radiation and nonradiation workers as discussed in Section 7.7 of the Generic SAR (Ref. 3).

7.7.1 RADIOLOGICAL MONITORING AND SURVEYS

The RBOF and RRF follow the SRS guidelines on radiological monitoring and surveys as described in WSRC Procedure Manual 5Q (Ref. 5). The RBOF and RRF are in accord with the SRS programs on radiological monitoring and surveys as described in Section 7.7 of the Generic SAR (Ref. 3), which includes the following topics:

- General requirements
- Radiation surveys
- Contamination surveys
- Area radiation monitors
- Airborne radioactivity monitoring

7.7.2 CONTAMINATION CONTROL

In general, engineered controls, as opposed to administrative controls, are the preferred option for contamination control. Airborne radioactivity is controlled at its source through the application of engineered controls and through the proper operation and maintenance of equipment and systems that can cause airborne radioactivity. Ventilation and containment systems are examples of engineered controls used to eliminate or limit the need for respiratory equipment.

In addition, eating, drinking, chewing, and smoking are prohibited inside radiological buffer areas, contamination areas, high contamination areas, and airborne radioactivity areas to minimize the chance of contaminated materials being ingested. Under certain circumstances, such as high risk of heat stress, drinking may be permitted inside the controlled area (Ref. 5) or from foot-operated water fountains inside the radiological buffer areas.

7.7.2.1 General

The RBOF and RRF limit radioactive contamination in accordance with the requirements specified in WSRC Procedure Manual 5Q (Ref. 5) and discussed in Section 7.7 of the Generic SAR (Ref. 3).

7.7.2.2 Ventilation Systems

The ventilation system for the RBOF and RRF serves to control the airborne radioactivity levels inside and outside the facility. The ventilation system is designed with a multiple-air-zone concept. Under this concept, ventilation air flows from areas with a low potential for contamination to areas with a higher potential for contamination. The facility is maintained at less than atmospheric pressure so that all building air will be filtered before being released. The basin areas are also kept at a lower pressure than the rest of the building, and a separate ventilation system supplies air to the control room, offices, and change room.

The exhaust system for the process vessels in the waste cell and the decontamination cell is similar to the building system but is separate and employs acid-resistant components.

Chapter 2 describes the ventilation systems and associated components for the RBOF and RRF.

7.7.2.3 Leak Detection

SFSD Maintenance follows the sitewide procedures found in WSRC Procedure Manual 9B (Ref. 9) to report leaks.

7.7.2.4 Contamination Control

The RBOF and RRF follow the SRS guidelines on contamination control as described in WSRC Procedure Manual 5Q (Ref. 5). The SRS Contamination Control Program is detailed in Section 7.7 of the Generic SAR (Ref. 3), which includes the following topics:

- Decontamination
- Protective clothing
- Work habits
- Documentation and trending

7.7.3 ASSOCIATED RECORDS/REPORTS

RBOF and RRF records are maintained in accordance with SRS document control procedures and the Sitewide Records Inventory and Disposition Schedule as explained in Section 7.7 of the Generic SAR and WSRC-IM-93-0060 (Ref. 3, 10).

7.7.4 METEOROLOGICAL DATA COLLECTION AND EVALUATION

WSRC-SCD-7 (Ref. 11) provides details of SRS meteorology programs applicable to the RBOF and RRF. WSRC-SCD-7 provides details on the following meteorologically related topics (including recurrence and severity):

- Hurricanes
- Tornadoes
- Ice and snow
- Thunderstorms
- Extreme winds
- Severe drought
- Extreme air pollution episodes

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7.8 RADIOLOGICAL PROTECTION INSTRUMENTATION

7.8.1 CRITERIA FOR SELECTION OF EQUIPMENT AND INSTRUMENTATION

The criteria for selection of equipment and instrumentation for the RBOF and RRF are consistent with the site program as detailed in Section 7.8 of the Generic SAR (Ref. 3).

7.8.2 CONTROL OF THE CALIBRATION PROCESS

Section 7.8 of the Generic SAR (Ref. 3) discusses control of the calibration process.

7.8.3 QUALITY ASSURANCE FOR CALIBRATION AND MAINTENANCE

Section 7.8 of the Generic SAR (Ref. 3) discusses quality assurance for calibration and maintenance.

7.8.4 TYPES OF DETECTORS AND MONITORS

Chapter 2 describes the detectors and monitors of the RBOF and RRF that form part of the facility Radiological Protection Program.

7.8.4.1 Air Monitors

The RBOF and RRF operate with one portable continuous air monitor for airborne contamination monitoring.

7.8.4.2 Gamma Monitors

A system of gamma monitors provides general low-level monitoring.

7.8.4.3 Nuclear Incident Monitors

Six nuclear incident monitors are installed in pairs at three locations. These monitors provide prompt evacuation alarms that alert facility personnel in the event of a criticality incident.

7.8.4.4 Other Radiation Detectors

One Kanne monitor is installed in the RBOF and RRF for the detection of airborne tritium. The Kanne monitor is available for use during cask venting and initial entry during unloading

operations. However, the normal practice for these operations is to provide full RCO coverage with hand held instruments. The RBOF and RRF are also equipped with an assortment of manually operated detectors for general survey of potentially contaminated surfaces and for measurement of radiation levels. In addition, PCM-1Bs are located at the radiological buffer area exit from the basin area and in the RRF Control Room.

7.8.5 CONTAMINATION CONTROL EQUIPMENT AND FACILITIES

Contamination is confined through sound process equipment specification and placement, coupled with high integrity cabinetry, that provide multiple confinement barriers to contamination problems throughout the process.

The filter-deionizer loop located in the decontamination cell is used to remove waterborne radioactive contamination from basin and canal water.

7.9 RADIOLOGICAL PROTECTION RECORDKEEPING

WSRC Procedure Manuals 5Q and 5Q1.1 (Ref. 5, 6) contain the prescribed practices for preparing and retaining radiologically related records at the RBOF and RRF. Section 7.9 of the Generic SAR (Ref. 3) details record keeping practices for the following topics:

- Inventory, retention, and disposition of records
- Reports
- Maintenance of plans and procedures

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7.10 OCCUPATIONAL RADIATION EXPOSURE

Radiological worker dose limits set by DOE are listed in Section 7.6 of the Generic SAR (Ref. 3) and provide upper bounds for exposure of operations personnel. The Generic SAR also lists the more aggressive DOE administrative control level of 2,000 mrem/yr per person established for all DOE activities. The radiological worker dose limits and DOE administrative control level are part of the SRS Radiological Control Program specified in WSRC Procedure Manual 5Q (Ref. 5).

Manual 5Q specifies that a facility administrative control level will be established and indicates that the facility administrative control level should be below 1,500 mrem/yr per person to be sufficiently challenging to meet the goals of Manual 5Q. The maximum individual annual exposures received by RBOF and RRF personnel in 1993 and 1994 were approximately 1,800 mrem and 372 mrem, respectively (Ref. 12). However, the maximum individual annual exposure for 1993 was assigned to an individual who was transferred to the RBOF/RRF during the course of the year and who had received approximately 300 to 400 mrem of the 1,800 mrem at the previous facility. Therefore, the maximum individual annual exposures actually resulting from RBOF and RRF operations for 1993 and 1994 were each less than 1,500 mrem.

The RBOF/RRF Facilities radiological summary for 1995 (Ref.13) and 1996 (Ref.14) provide information relative to the facilities radiological goals, and comparison to other SFS division facilities and the site. In 1995, the quarterly average for RBOF/RRF was 300 mrem (deep dose), as compared to the site average of 62 mrem average quarter for the same time period. In 1996, the quarterly average for RBOF/RRF was 576 mrem (deep dose), as compared to the site average of 57 mrem average quarter for the same time period. The last quarter of 1996 was the highest reported exposure, due to increased fuel receipts into the facility and the establishment of additional shifts to handle the increased work load.

Personnel exposures in the RBOF and RRF are controlled in accordance with the requirements specified in Manual 5Q and indicated in Section 7.6 of the Generic SAR and are maintained ALARA.

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 8

HAZARDOUS MATERIAL PROTECTION PROGRAM

AUGUST 1997

UNCLASSIFIED DOES NOT CONTAIN UNCLASSIFIED CONTROLLED NUCLEAR INFORMATION	
ADC & Reviewing Official:	<u>J. Dewes</u>
J.N. DEWES, Manager, Regulatory Programs	
Date:	<u>9/23/97</u>

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

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SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

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ACRONYMS

ALARA	As Low As Reasonably Achievable
DOE	Department of Energy
HAZCOM	Hazards Communication
IH	Industrial Hygiene
IH&RP	Industrial Hygiene and Respiratory Protection
RBOF	Receiving Basin for Offsite Fuels
RRF	Resin Regeneration Facility
SAR	Safety Analysis Report
SRS	Savannah River Site
WSRC	Westinghouse Savannah River Company

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8.0 HAZARDOUS MATERIAL PROTECTION PROGRAM

8.1 INTRODUCTION

8.1.1 OBJECTIVE

The purpose of this chapter is to provide information on hazardous material protection that satisfies the requirements of Department of Energy (DOE) Order 5480.23, paragraph 8.b.(3)(j), as amplified in Attachment 1, paragraphs 4.f.(3)(d)10 and 4.f.(3)(d)11a, of the Order (Topic 10) (Ref. 1). This chapter also includes information, as applicable, that partially satisfies the requirements of paragraphs 8.b.(3)(b), (f), and (u) of the Order, which are addressed in each chapter of the Safety Analysis Report (SAR) as necessary to provide explicit correlation to topics (Ref. 2).

8.1.2 SCOPE

This chapter summarizes the provisions for the protection of workers, the public, and the environment from nonradioactive hazardous materials associated with the Receiving Basin for Offsite Fuels (RBOF) and the Resin Regeneration Facility (RRF), which are part of the Spent Fuel Storage Division at the Savannah River Site (SRS). The scope of this chapter includes the following:

- Description of the overall Hazardous Material Protection Program and organization
- Description of the hazardous material As Low As Reasonably Achievable (ALARA) policy and program
- Description of hazardous material exposure control, including identification of hazardous material, administrative limits, occupational medical programs, and respiratory protection
- Identification of the hazardous material monitoring conducted to protect workers, the public, and the environment
- Discussion of hazardous material protection instrumentation
- Description of the plans and procedures for maintaining records of hazardous materials, Hazards Communications (HAZCOM), and occupational exposures

Chapter 8 of the Generic SAR (Ref. 3) describes the Hazardous Material Protection Program at SRS. This chapter presents facility-specific information, referencing the Generic SAR, as well as other chapters, as appropriate.

Chapter 3 provides a comprehensive evaluation of the nonradioactive hazardous materials that are associated with the RBOF and RRF. The hazards analysis evaluates the consequences of hazardous chemical spills for the RBOF and RRF due to probable causes such as the following:

- Material failure
- Mechanical damage
- Dropping of a vessel during movement
- Equipment malfunction
- Worker error
- Overfilling
- Overpressurization
- Corrosion
- Natural phenomena
- Extrinsic events

8.2 REQUIREMENTS

Section 8.2 of the Generic SAR (Ref. 3) lists the DOE Orders that apply to the site Industrial Hygiene (IH) Program and discusses the Westinghouse Savannah River Company (WSRC) documents that govern the IH Program, including the Hazardous Material Protection Program. The Hazardous Material Protection Program for the RBOF and RRF is administered through the site IH Program delineated in WSRC Procedure Manual 4Q (Ref. 4).

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8.3 INDUSTRIAL HYGIENE ORGANIZATION

The site IH Program is managed by the Industrial Hygiene and Respiratory Protection (IH&RP) Section of the Occupational Safety and Hygiene Department of the Environmental, Safety, Health, and Quality Assurance Division. The RBOF and RRF share one industrial hygienist and two industrial technicians with other H-Area facilities. In addition, facility engineers are responsible for addressing IH issues associated with the RBOF and RRF, and the HAZCOM Program Coordinator is responsible for addressing HAZCOM issues for the RBOF and RRF.

Section 8.3 of the Generic SAR (Ref. 3) describes the responsibilities of all levels of the WSRC organization to implement the IH Program.

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8.4 ALARA POLICY AND PROGRAM

An ALARA policy is an inherent part of the WSRC IH policy and program at SRS. The RBOF and RRF use the site generic ALARA policy and program as described in Section 8.4 of the Generic SAR (Ref. 3). There are no facility-specific requirements associated with the ALARA policy or program.

8.4.1 INDUSTRIAL HYGIENE POLICY

Section 8.4 of the Generic SAR (Ref. 3) describes the WSRC IH policy, including the Hazardous Material Protection Program, and the responsibilities of the WSRC organization to comply with the IH policy.

8.4.2 INDUSTRIAL HYGIENE PROGRAM

The site IH Program is implemented through WSRC Procedure Manual 4Q (Ref. 4) and is managed by IH&RP, as described in Section 8.4 of the Generic SAR (Ref. 3).

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8.5 HAZARDOUS MATERIAL TRAINING

Section 8.5 of the Generic SAR (Ref. 3) specifies the site IH training requirements for IH&RP personnel and other WSRC personnel and subcontractors. HAZCOM training for the RBOF and RRF provides information to personnel about the specific hazardous materials that they may be exposed to during the normal performance of their work and the appropriate protective measures that they should take.

8.5.1 INDUSTRIAL HYGIENE AND RESPIRATORY PROTECTION STAFF

There are no facility-specific training requirements for the IH&RP staff. Section 8.5 of the Generic SAR (Ref. 3) specifies the IH training requirements and guidelines for the various IH&RP staff positions.

8.5.2 WSRC PERSONNEL AND SUBCONTRACTORS

There are no facility-specific training requirements for WSRC personnel and subcontractors. Section 8.5 of the Generic SAR (Ref. 3) lists the site IH training requirements, including hazardous material training requirements for WSRC personnel and subcontractors.

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8.6 HAZARDOUS MATERIAL EXPOSURE CONTROL

8.6.1 HAZARDOUS MATERIAL IDENTIFICATION PROGRAM

Section 8.6 of the Generic SAR (Ref. 3) specifies the requirements of the IH Program to perform and document periodic hazards assessments, to maintain hazards assessment and control records, and to enter and maintain these records on a sitewide data management system. The Generic SAR section also discusses the methods for identifying and evaluating health and chemical hazards in addition to determining the adequacy of, and the need for, hazard prevention and control measures. The RBOF and RRF satisfy applicable site requirements.

Chapter 3 presents the chemical hazards resulting from postulated accidents in the RBOF and RRF.

8.6.2 ADMINISTRATIVE LIMITS

WSRC uses the specified industrial and governmental regulatory limits (i.e., permissible exposure limits and threshold limit values) implemented through WSRC Procedure Manual 4Q (Ref. 4) for their nonradioactive hazardous material administrative control levels and exposure limits. The RBOF and RRF satisfy applicable site requirements.

8.6.3 OCCUPATIONAL MEDICAL PROGRAM

Section 8.6 of the Generic SAR (Ref. 3) states the WSRC medical policy, associated program requirements, and the IH programmatic responsibilities of the Medical Department. The RBOF and RRF satisfy applicable site requirements.

8.6.4 RESPIRATORY PROTECTION PROGRAM

Section 8.6 of the Generic SAR (Ref. 3) describes the Respiratory Protection Program. The Manager of IH&RP has overall responsibility for the Respiratory Protection Program.

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8.7 HAZARDOUS MATERIAL MONITORING

8.7.1 HAZARDOUS MATERIAL MONITORING

Monitoring of nonradioactive hazardous materials for the RBOF and RRF consists of air monitoring and is accomplished by periodic IH surveys conducted by the IH&RP staff. Nonradioactive hazardous material survey frequencies are dependent upon the type of hazardous materials involved and previous survey results.

Section 8.7 of the Generic SAR (Ref. 3) provides additional information on sitewide aspects of nonradioactive hazardous material monitoring. Section 7.7 discusses the monitoring of radiological hazards for the RBOF and RRF.

8.7.2 AIR MONITORING

This section discusses the nonradioactive hazardous material air sampling and monitoring programs conducted for the RBOF and RRF. There are no fixed or continuous nonradioactive hazardous material air monitors provided in the facility.

8.7.2.1 Air Monitoring Inside the Facility

The site IH Program specifies the requirements for workplace air sampling and data analysis. Section 8.7 of the Generic SAR (Ref. 3) discusses the specific requirements. The RBOF and RRF satisfy applicable site requirements.

Portable hand-held equipment is used in the RBOF and RRF for monitoring nonradioactive hazardous material. The IH&RP staff conducts IH surveys to obtain air samples inside the RBOF and RRF. The IH survey procedures specify air sampling and equipment requirements. Samples are analyzed at the site laboratories. The results are then evaluated and documented on IH data forms by the H-Area Industrial Hygienist. Samples of breathing air are taken as required by WSRC Procedure Manual 4Q (Ref. 4). Records and reports are generated by IH&RP to inform RBOF and RRF management of the results and any recommendations.

8.7.2.2 Air Monitoring Outside the Facility

Nonradioactive hazardous material air monitoring conducted outside the RBOF and RRF is part of the site Nonradioactive Hazardous Material Air Monitoring Program and is not specific to the RBOF and RRF.

8.7.3 HAZARDS PREVENTION AND CONTROL

The site IH Program establishes the guidance for identifying and recommending effective engineering, work practice, and administrative controls to reduce employee exposures to occupational hazards. Section 8.7 of the Generic SAR (Ref. 3) describes the guidance and requirements for the control of hazardous materials. The RBOF and RRF satisfy applicable site requirements.

8.8 HAZARDOUS MATERIAL PROTECTION INSTRUMENTATION

Monitoring of nonradioactive hazardous materials for the RBOF and RRF consists of air monitoring as discussed in Section 8.7. As indicated in that section, there are no fixed or continuous nonradioactive hazardous material air monitors provided in the facility. The IH&RP staff uses portable hand-held equipment to conduct IH surveys within the facility. The IH survey procedures specify equipment requirements.

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8.9 HAZARDOUS MATERIAL PROTECTION RECORDKEEPING

Section 8.9 of the Generic SAR (Ref. 3) addresses recordkeeping requirements for hazardous material protection information applicable to SRS facilities. The RBOF and RRF satisfy applicable site requirements.

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8.10 HAZARDS COMMUNICATION PROGRAM

The site IH Program establishes the site HAZCOM Program to implement the provisions of the Occupational Safety and Health Act standard for communicating chemical hazards to employees at SRS. HAZCOM training provides facility-specific information to facility personnel on the specific chemical hazards in their work area. Section 8.10 of the Generic SAR (Ref. 3) describes elements of the program.

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8.11 OCCUPATIONAL CHEMICAL EXPOSURES

The predicted exposure levels of facility workers during normal operations in the RBOF and RRF from nonradioactive hazardous materials are expected to be negligible based on the hazards analysis presented in Chapter 3.

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8.12 REFERENCES

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 9

RADIOACTIVE AND HAZARDOUS MATERIAL WASTE MANAGEMENT

AUGUST 1997

UNCLASSIFIED DOES NOT CONTAIN UNCLASSIFIED CONTROLLED NUCLEAR INFORMATION	
ADC & Reviewing Official:	<u>J.N. Dewes</u>
J.N. DEWES, Manager, Regulatory Programs	
Date:	<u>9/23/97</u>

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

SAVANNAH RIVER SITE

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

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SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

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ACRONYMS

ALARA	As Low As Reasonably Achievable
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
DOE	Department of Energy
EPA	Environmental Protection Agency
ES&H	Environmental, Safety, and Health
NOP	Normal Operating Procedure
RBOF	Receiving Basin for Offsite Fuels
RC&HP	Radiological Control and Health Physics
RCRA	Resource Conservation and Recovery Act
RRF	Resin Regeneration Facility
SCDHEC	South Carolina Department of Health and Environmental Control
SFSD	Spent Fuel Storage Division
SRS	Savannah River Site
TSD	Treatment, Storage, or Disposal
WAC	Waste Acceptance Criteria
WITS	Waste Information Tracking System
WSRC	Westinghouse Savannah River Company

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9.0 RADIOACTIVE AND HAZARDOUS MATERIAL WASTE MANAGEMENT

9.1 INTRODUCTION

The Receiving Basin for Offsite Fuels (RBOF) and the Resin Regeneration Facility (RRF) consist of process areas in Buildings 244-H and 245-H. The RBOF and RRF perform two major functions for the Department of Energy (DOE) at the Savannah River Site (SRS). The primary function revolves around the receipt and storage of irradiated nuclear fuel elements in Building 244-H. The RBOF and RRF are capable of processing and storing the fuel elements in underwater basins. Fuel elements are prepared for shipment to either H or F Canyon, the Savannah River Technology Center, the K- or L-Area disassembly basins, or an offsite location.

Resin regeneration constitutes the second function. The resin used in deionizers to remove radioactive and nonradioactive ions from water is received from the Reactor Areas in portable deionizers, which are permanently mounted on truck trailers. The RBOF and RRF also regenerate the resin used in its own filter-deionizer system. The filter-deionizer system reduces the level of radioactivity in, and removes the impurities from, the water used in the basins of stored fuel elements to maintain water quality and thereby minimize corrosion, contamination, and personnel exposure.

These tasks involve conducting the following primary activities:

- Cask receipt
- Cask washing
- Cask steam cooling
- Fuel unloading
- Fuel transfer
- Fuel disassembly
- Fuel repackaging
- Fuel shipping
- Basin water purification
- Resin regeneration
- Liquid waste handling

9.1.1 OBJECTIVE

This chapter provides information relating to radioactive and hazardous waste management at the RBOF and RRF that satisfies the requirements of DOE Order 5480.23, paragraph 8.b(3)(g), as

amplified in Attachment 1, paragraph 4.f.(3)(d)7, of the Order (Topic 7) (Ref. 1). This chapter documents estimates of the quantity and form of radioactive and hazardous wastes generated incidental to the mission of the RBOF and RRF, as well as equipment, programs, and plans for the management of these wastes. Additionally, this chapter documents the solid, liquid, gaseous, and mixed radioactive wastes and toxic chemical waste forms and provides information regarding compliance with the requirements of DOE Order 5820.2A (Ref. 2).

9.1.2 SCOPE

This chapter describes the program and organization for radioactive and hazardous waste management at the RBOF and RRF.

9.2 REQUIREMENTS

Radioactive, hazardous, and mixed wastes generated at the RBOF and RRF are managed under an extensive network of federal and state laws and regulations and DOE Orders and requirements. The policy of the RBOF and RRF is to comply with the letter and spirit of federal, state, and local requirements concerning environmental protection and waste management. SRS's waste management objectives are to comply with the applicable federal and state regulations, DOE Orders, and Westinghouse Savannah River Company (WSRC) policies to minimize effects on the environment and to minimize waste generation as much as technically and economically practicable. In the area of waste management, the primary objective of the RBOF and RRF is to minimize impacts to the public and the environment, keeping all impacts As Low As Reasonably Achievable (ALARA) below allowable limits. Waste management operations at the RBOF and RRF are also conducted in a manner that minimizes the impact to workers.

SRS's waste management practices are regulated by DOE, the Environmental Protection Agency (EPA), and the South Carolina Department of Health and Environmental Control (SCDHEC). The Resource Conservation and Recovery Act (RCRA) provides a comprehensive EPA regulatory program for hazardous waste management and a state regulatory program for nonhazardous waste management. SCDHEC has been granted authority by the EPA to conduct most elements of the RCRA Hazardous Waste Program in South Carolina.

Various environmental statutes, laws, or regulations emanating from Congress or through executive orders from the President are implemented at DOE sites through DOE Orders or Directives. These Orders and Directives set policies and define responsibilities for DOE's site and field personnel and its contractors. The Health and Safety Department provides assistance to line managers for both radiological and nonradiological occupational safety. The Environmental Protection Department is responsible for assisting the line managers in protecting the environment from operations at the RBOF and RRF.

WSRC Procedure Manuals 3Q, 5Q, and 1S (Ref. 3, 4, 5) are site-level documents that implement DOE Environmental, Safety, and Health (ES&H) policies and guidelines for the safe management and disposal of radioactive and hazardous wastes at SRS. WSRC Procedure Manuals RD-1 and R14.1 (Ref. 6, 7) are division-level documents that implement related requirements at Spent Fuel Storage Division (SFSD) facilities. Facility-specific procedures have been written to address the applicable requirements of these manuals.

DOE's waste management policy is outlined in DOE Order 5820.2A (Ref. 2). The policy states that any radioactive, hazardous, or mixed wastes shall be managed in a manner that ensures protection of the environment and the health and safety of the public, DOE employees, and contractor employees. The generation, treatment, storage, transportation, and/or disposal of radioactive, hazardous, or mixed waste shall be accomplished in a manner that minimizes the generation of such wastes; complies with all federal, state, and local ES&H laws and regulations; and complies with DOE requirements. DOE Order 5820.2A requirements apply to the

management of low-level radioactive wastes and mixed waste. The Order states that mixed waste "... shall also be regulated by the appropriate regional authorities under the RCRA." A section of this Order deals with the Waste Acceptance Criteria (WAC), a topic covered in Manual 1S.

DOE Order 5440.1E (Ref. 8) establishes the internal DOE responsibilities and procedures for implementing the National Environmental Policy Act. The RBOF and RRF implement the requirements of Manual 3Q (Ref. 3) to achieve satisfactory results for each of these areas. DOE Order 5480.1B (Ref. 9) establishes the ES&H Program for DOE operations with DOE Order 5400.1 (Ref. 10) more specifically defining the environmental protection requirements. DOE Order 5480.4 (Ref. 11) lists applicable environmental protection, safety, and health protection standards.

DOE Order 5400.1 establishes a policy for DOE to conduct all its operations in compliance with applicable environmental statutes, regulations, and standards. DOE Order 5400.2A (Ref. 12) sets forth DOE policy, direction, and procedures for coordination of significant environmental compliance issues to ensure timely development and consistent application of DOE environmental policy and guidance. DOE Orders 5484.1, 5482.1B, and 5700.6C (Ref. 13, 14, 15) address associated reporting requirements, appraisal programs, and quality assurance, respectively. SRS requirements in these areas are delineated in WSRC Procedure Manuals 1B, 1Q, 3Q, 11Q, and 1S (Ref. 16, 17, 3, 18, 5). SFSD requirements in these areas are delineated in Manuals RD-1 and R14.1 (Ref. 6,7).

DOE Order 5400.4 (Ref. 19) establishes the requirements of the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA). DOE Order 5400.4 also establishes the instructions for implementing the DOE CERCLA Program, defines actions to identify and evaluate inactive hazardous waste sites at DOE installations, and directs the custodian to take remedial actions where necessary to improve control of hazardous substance migration from the site. DOE Order 5400.5 (Ref. 20) establishes the standards and requirements for DOE operations with respect to protection of the public and the environment against undue risk from radiation. These requirements are implemented in Manual 3Q (Ref. 3) along with other manuals discussed in this section.

DOE Order 5480.11 (Ref. 21) establishes DOE-radiation protection standards and program requirements with respect to the protection of the worker from ionizing radiation. DOE Order 5480.11 requirements are implemented through the SRS ALARA programs that are detailed in Manual 5Q (Ref. 4). Chapter 7 details the implementation of the requirements of DOE Order 5480.11 and provides descriptions of the equipment and procedures used for radiation protection of RBOF and RRF workers.

DOE Order 5483.1A (Ref. 22) establishes the requirements and procedures for providing safety and health protection for DOE contractor employees at government-owned, contractor-operated facilities that is consistent with the protection afforded private industry employees by Occupational Safety and Health Act standards. These requirements are implemented by WSRC Procedure Manuals 8Q and 20Q (Ref. 23, 24). DOE Order 5480.10 (Ref. 25) establishes the

requirements and guidelines applicable to DOE operations for maintaining an effective Industrial Hygiene Program to preserve employee health and well-being. WSRC Procedure Manual 4Q (Ref. 26) contains the procedures implemented to achieve satisfactory results for each of these areas.

The waste generator requirements of Title 40 Code of Federal Regulations (CFR) Part 262 (Ref. 27) reference the applicable Department of Transportation regulations under 49 CFR 172-179 (Ref. 28), relative to hazardous waste classifications, packaging, container labeling and marking, and vehicle placarding while waste is in transit. Container selection, labeling, and marking also are required by the EPA while the waste is being accumulated or stored before treatment or disposal. DOE Order 5480.3 (Ref. 29) establishes requirements for packaging and transporting hazardous materials, including radioactive hazardous substances and hazardous wastes. The requirements of WSRC Procedure Manual 19Q (Ref. 30) are implemented to satisfy these requirements.

WSRC complies with applicable environmental regulations by implementing DOE Orders and by issuing policies, procedures, guidance manuals, or, in some cases, a special procedure specific to a certain activity or operation. Manuals 3Q and 1S (Ref. 3,5) are used to implement DOE Orders, as well as Federal and State Regulations in the area of waste management. Manual 3Q contains site procedures that should be consulted when dealing with hazardous, radioactive mixed waste, and air and liquid effluents (waste). Manual 1S provides written guidance to SRS environmental personnel or waste generators to characterize, manage, store, treat, or dispose of such waste according to all regulatory requirements, WSRC policies, and DOE Orders. The manual provides a summary of the following:

- EPA and SCDHEC regulations applicable to hazardous waste determination
- Selection of compatible containers
- Management of containerized waste in accumulation areas and storage facilities
- Preparation for shipment to onsite or offsite Treatment, Storage, or Disposal (TSD) facilities

Currently, SRS has in place a Standards Requirements Identification Documents (SRIDS) program which identifies the applicable requirements and determines whether SRS procedures adequately implement the requirements. The vast majority of requirements are correctly implemented, but several Orders have been found not to be in complete compliance. SRIDS not only identifies the noncompliant item but also sets up a database with commitment dates for compliance.

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9.3 RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT PROGRAM AND ORGANIZATION

The Waste Management Program at the RBOF and RRF consists of operating procedures and a waste minimization plan to ensure that the waste generated in the facility and the waste leaving the facility are in accordance with the WAC of the applicable SRS solid waste management TSD facility and that environmental concerns are addressed (Ref. 31). Liquid waste, depending on its radioactive content, is transferred either to the Building 211-H Outside Facilities or to the H-Area tank farm.

It is the policy of WSRC to manage radioactive, nonradioactive, hazardous, and mixed wastes to minimize the risk to humans and the environment over the effective lifetime of these wastes. This policy is accomplished by a combination of building and equipment design and construction as well as facility operation. As discussed previously, operation of the RBOF and RRF is consistent with national, state, and procedural requirements (including DOE Orders 5820.2A and 5400.1 [Ref. 2, 10]) for waste management and environmental protection.

The operations and features of waste confinement and management facilities, systems, and equipment are also designed and controlled to comply with DOE and WSRC site policies governing the management of hazardous (including radioactive) wastes. The policies include the following (Ref. 32):

- Management of the Quantities of Waste Produced: The generation of hazardous and environmentally polluting wastes (including radioactive, hazardous, and mixed wastes) in gaseous (or airborne particulate), liquid, and solid forms shall be limited to quantities that are as low as economically practical within the constraints of available and feasible technologies and methods.
- Management of Hazardous Waste Releases: The goal for the design and operation of systems used to confine or otherwise prevent the release of hazardous materials shall be that no unmanaged releases of hazardous material will be made to the environment or to the work place.
- Characterization, Classification, Certification, and Segregation of Hazardous Wastes: The design and operation of facilities and systems producing hazardous wastes shall provide for characterization, classification, certification, and segregation of the wastes in an effort to enhance public and worker health and safety as well as provide for protection of the environment during handling, transportation, treatment, storage, and final disposal.

All RBOF and RRF waste materials (excluding sanitary waste) and contaminated equipment are disposed of onsite. Sanitary waste is disposed of offsite. At SRS, effluent releases to the environment are controlled by efficient removal systems operated in a manner that maintains radiological releases below established release guides and nonradiological releases below the outfall permit levels. The release guides list values for releases of radioactive material that are

considered achievable with good operating practice and that are well below federal guidelines. The release guides set goals for atmospheric releases and for liquid releases to streams. The guides are not specific to the RBOF and RRF but cover the entire H Area or both F and H Areas.

Each hazardous and radioactive waste generator that delivers waste to SRS TSD facilities is required to implement a waste certification program that is described in a waste certification plan. This program ensures that any facility receiving waste can be assured that the waste meets the WAC for the facility. In addition, the waste certification program describes the methods and systems that enable personnel to conform to the standards of WSRC Procedure Manual 1S (Ref. 5).

Waste certification ensures that waste has been properly identified, characterized, segregated, packaged, and shipped to the appropriate receiving facility according to that receiving facility's WAC. Under this program, each waste generator designates a Generator Certification Official to administer the waste generator's certification program and to ensure that the waste generator's waste management programs implement and document controls that meet applicable WAC.

The waste certification program is used by WSRC waste generators to identify, characterize, and package waste in compliance with approved procedures and administrative controls that are documented in the facility's waste certification plan. The program ensures continuity within the site and between various generators. The program also ensures compliance with DOE requirements at the approved facilities. In the WAC 3.0 Series of Manual 1S, responsibilities and operations have been detailed to allow the movement and storage of waste at the RBOF and RRF. DOE Order 5820.2A (Ref. 2) requires that all radioactive waste generated onsite be characterized and that each generator develop a waste certification program.

The Waste Information Tracking System (WITS) is used by site generators and TSD facilities to document and track radioactive waste from its generation to its ultimate storage or disposal. Information relating to the waste characterization is entered into WITS by the generator, and a shipping manifest is printed. This manifest travels with the waste to the receiving facility. The receiving facility performs appropriate receipt inspection activities, then acknowledges receipt of the waste on WITS. In this manner, the location of the waste can be determined at any time

The site-level organizational responsibilities associated with transferring wastes to TSD facilities are described below. Each area of responsibility is charged with assignments which will ensure that DOE Orders are satisfied. The following descriptions are not the total list of the responsibilities but are a summary of them:

- The Environmental Protection Department is responsible for providing oversight to the SRS Environmental Program and for representing WSRC, in conjunction with the Office of General Counsel, on behalf of DOE when interacting with federal, state, and local agencies regarding environmental compliance and permits.
- The Office of General Counsel is responsible for reviewing and interpreting environmental laws and regulations.

- Radiological Control and Health Physics (RC&HP) is responsible for ensuring that radioactive materials shipped onsite meet the requirements of WSRC Procedure Manual 5Q (Ref. 4), which describes the Radiological Protection Program at SRS as specified in DOE Order 5480.11 (Ref. 21).
- The WSRC Transportation Department is responsible for ensuring that waste is transported in a manner that protects operations personnel and prevents the release of radioactive or chemical contamination on site roadways in accordance with WSRC Procedure Manual 19Q (Ref. 30).
- The Generator Cognizant Quality Function and Solid Waste Quality Assurance Sections of the Quality Assurance Department are responsible for performing audits or assessments, as needed, of the waste generator's quality assurance program associated with waste certification.
- The Solid Waste Management Department is responsible for managing the overall solid waste program at SRS. Duties include management of facilities in compliance with applicable laws, regulations, and permits for the protection of the public and the environment and with applicable DOE Orders and Directives.
- Solid Waste Engineering is responsible for developing, maintaining, reviewing, and revising the WAC and for ensuring that procedures comply with DOE and state regulations.
- Solid Waste Operations is responsible for accepting, treating, storing and disposing of SRS-generated radioactive and hazardous waste.
- Solid Waste Maintenance is responsible for operating the vehicles and equipment that are used in transporting, storing, or disposing of waste within SRS solid waste management facilities.

SFSD Environmental Compliance Engineering, Reactor Waste Management, and Reactor Quality Assurance have related division-level responsibilities as specified in WSRC Procedure Manuals RD-1 and R14.1 (Ref. 6, 7). In addition, waste management operations at the RBOF and RRF are governed by Technical Safety Requirements (Ref. 33), as well as Standing Orders and Normal Operating Procedures (NOPs). Administrative and engineered controls at the RBOF and RRF are discussed in Chapters 3, 5, and 11.

Wastes will not remain in the RBOF or RRF during the post-closure period. Chapter 16 discusses decontamination and decommissioning. This chapter does not address post-closure waste containment.

9.3.1 WASTE MINIMIZATION

The RBOF and RRF operate in terms of priority lists for both waste minimization techniques and targeted waste streams. Progression through the priority lists is from top to bottom. A leveling off of an improvement factor initiates a discussion between the Basin Facility Manager and the

SFSD Waste Coordinator as to the need to either attack a waste stream through another waste minimization technique or focus efforts on another waste stream.

The primary objective of the SFSD Waste Minimization Plan (Ref. 31) is to reduce waste generation to as low a level as feasibly and practically possible. The following elements are key to the success of the program:

- Waste handling is included in the RBOF Building General On-the-Job Training so that all personnel qualified in this area are allowed to perform waste handling activities.
- Liquid Waste Handling System Training is given to all operations personnel as a supplement to RBOF Building General On-the Job Training.
- Operations personnel are required to attend annual waste minimization training/awareness courses.
- Waste minimization and source reduction are considerations for every job performed at the facility, either through job preplans, proceduralized information, or personnel awareness.

Waste minimization performance is tracked using facility operating procedures, procurement control, and waste reporting.

The SFSD Waste Minimization Plan is revised annually to issue goals designed to assist in the achievement of waste minimization objectives.

9.3.1.1 Prioritized List of Waste Minimization Techniques

The following is a prioritized list of waste minimization techniques for the RBOF and RRF; these techniques are discussed in the following sections:

- Source reduction
- Recycle/reuse
- Treatment
- Disposal

SOURCE REDUCTION

Source reduction involves reducing the generation of waste at the source. This technique includes the following:

- Process modifications
- Material substitutions

- Procedure changes
- Procurement controls
- Inventory control
- Improved housekeeping
- Preventive maintenance
- Redesign of products

RECYCLE/REUSE

Recycle/reuse involves using materials to eliminate the need for declaring such materials as waste. This technique includes the following:

- Office waste recycling
- Product recovery
- Energy recovery
- Salvage sales
- Waste exchanges
- Waste segregation/reclamation

TREATMENT

Treatment involves reducing the volume, toxicity, and/or mobility of a waste stream. This technique includes the following:

- Waste segregation
- Incineration
- Evaporation
- Compaction
- Stabilization
- Decontamination
- Neutralization

DISPOSAL

Disposal involves disposing of waste in an environmentally safe and economical manner.

9.3.1.2 Prioritized List of Targeted Waste Streams

The following is a prioritized list of targeted waste streams for the RBOF and RRF:

- Nonradioactive solid waste
- Low-level radioactive solid waste
- Radioactive Liquid waste
- Intermediate-level radioactive solid waste

Each of these waste streams is discussed in Section 9.4.2, along with other typical waste streams.

9.3.1.3 Prioritized List of Process Waste Assessments

The RBOF and RRF are working in terms of the following prioritized list of waste streams requiring process waste assessments:

- Radioactive liquid waste
- Low-level waste
- Intermediate-level waste

9.3.2 ORGANIZATION

The RBOF and RRF are part of SFSD. The structure and interdisciplinary functions of the organizations within SFSD are described in WSRC Procedure Manual RD-1 (Ref. 6). The organizations involved specifically with division-level waste management operations are addressed in WSRC Procedure Manual R14.1 (Ref. 7).

The SFSD Chemical and Waste Coordinators are assigned by the SFSD Manager. The Chemical Coordinator oversees chemical usage and storage activities within SFSD facilities and is responsible for facilitating the disposal of excess chemicals. The Waste Coordinator directs waste packaging and storage activities within SFSD facilities and is responsible for ensuring that excess chemicals declared to be waste are disposed of properly.

The Basin Facility Manager is responsible for operating and maintaining the RBOF and RRF in accordance with site, state, and federal environmental regulations and DOE Orders. In addition, the Basin Facility Manager is responsible for implementing SFSD policies and programs at the facility level.

The RBOF and RRF, as part of SFSD, receive waste management support from several WSRC organizations that include, but are not limited to, the following:

- Waste Management
- Quality Assurance
- Facilities Support
- SFSD Environmental Compliance Engineering
- RC&HP
- Analytical Laboratories

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9.4 RADIOACTIVE, MIXED, AND HAZARDOUS WASTE STREAMS AND SOURCES

9.4.1 WASTE MANAGEMENT PROCESS

Chapter 11 indicates that normal RBOF/RRF operations, which include facility waste management operations, are conducted in accordance with RBOF/RRF Standing Orders and NOPs, supplemented by special procedures as appropriate. NOPs associated with facility waste management operations address, but are not limited to, the following activities:

- Waste handling
- Waste minimization
- Waste sampling
- Waste treatment
- Waste transfer

9.4.2 WASTE SOURCES AND CHARACTERISTICS

9.4.2.1 Nonradioactive Solid Waste

Nonradioactive solid waste includes paper waste (e.g., cups, towels, bags, and boxes) and garbage in the form of metal, cans, cleaning type trash, mops, and plastic bags that are generated in the kitchen, offices, and bathrooms. Waste that meets the criteria of the “clean” waste program also falls into this category.

9.4.2.2 Low-Level Radioactive Solid Waste

Low-level radioactive solid waste is made up of spent resin beds from deionizers and process waste such as filters, tools, plastic suits, rubber gloves, and spray cans. The only waste generation baselines determined to date are for low-level wastes and are in terms of cask/fuel handling operations because fluctuations in the amount of waste produced at the RBOF and RRF are linked to this activity.

9.4.2.3 Intermediate-Level Radioactive Solid Waste

Intermediate-level radioactive solid waste is composed almost entirely of scrap metal from RBOF basin work.

9.4.2.4 Radioactive Liquid Waste

Liquid waste streams are those generated from work on filter-deionizer system flushes, regeneration of resin from portable or RBOF/RRF deionizers, and cask/fuel handling activities (e.g., decontamination, basin purges). A program has been established (Ref. 36) to provide guidance to operations staff on radioactive liquid waste transfers out of the facility.

9.4.2.5 Gaseous Waste

Waste generated in the facility consists of gaseous radionuclides and airborne radioactive particulates. The ventilation system is used to remove the airborne radioactive particulates. The gaseous radionuclides are vented and monitored through the ventilation stacks.

9.4.2.6 Radioactive Mixed Waste

Radioactive mixed waste is not routinely generated by RBOF and RRF operations, it is handled as specified in the RBOF Waste Certification Plan (Ref. 34).

9.4.2.7 Hazardous Chemical Waste

Hazardous chemicals used in the RBOF and RRF include sodium hydroxide, nitric acid, and sodium nitrite as indicated by the hazards analysis in Chapter 3. Used chemical waste is usually collected in waste tank EP38. EP-38 is part of waste stream described in section 9.4.2.4. Non-routine hazardous chemical waste is handled in accordance with Manuals 3Q and 1S (Ref. 3,5) procedures.

9.4.3 WASTE HANDLING AND/OR TREATMENT SYSTEMS**9.4.3.1 Nonradioactive Solid Waste**

Nonradioactive solid wastes, including paper products, discarded office supplies, and lunchroom vending machine wastes, are collected and disposed of in an offsite sanitary landfill.

9.4.3.2 Low-Level Radioactive Solid Waste

Anticontamination clothing is usually sent offsite to be cleaned and returned to service. Radiological protective clothing that is no longer usable is packaged and sent to the burial grounds for disposal according to procedure.

9.4.3.3 Intermediate-Level Radioactive Solid Waste

Intermediate-level radioactive solid waste is transported to the solid waste management facilities for disposal.

9.4.3.4 Liquid Waste

Liquid wastes result from cask/fuel processing at the RBOF and from regeneration of deionizer resin beds at the RRF. Radiation monitoring is performed to determine if the waste may be sent to the Building 211-H Outside Facilities or if the waste must be sent to the H-Area tank farm. Sanitary waste from the RBOF and RRF is pumped to the 200-Area sanitary waste treatment area where it is monitored.

Solutions generated during cask decontamination operations are transferred to a waste tank in the RBOF or RRF and then either to feed tank 717 for the general purpose evaporator at the Building 211-H Outside Facilities or to tank 21 or 23 at the H-Area tank farm. A sample bottle of the waste solution is monitored to ensure it meets radiological and organic limits before the waste can be transferred to the general purpose evaporator (Ref. 35).

9.4.3.5 Gaseous Waste

Gaseous effluents are released through one of two facility stacks. Basin areas, the control room, and radiological portions of the Radiological Control Operations office and change room section are exhausted through the building exhaust filters before being discharged to the atmosphere through the 5-foot-diameter, 53-foot-high main building stack; the building exhaust filters in this system are preceded by roughing filters to extend the serviceability of the building exhaust filters. Process tanks are vented through the vessel vent exhaust filters in the vessel vent system before discharging to the atmosphere through the vessel vent stack.

Chapter 2 provides additional information concerning the building exhaust system, vessel vent system, and related components.

9.4.3.6 Radioactive Mixed Waste

There is no radioactive mixed waste produced by RBOF and RRF operations as specified in the RBOF Waste Certification Plan (Ref. 34).

9.4.3.7 Hazardous Chemical Waste

The hazardous chemicals listed in Section 9.4.2.7 are collected in waste tank EP38. This tank is used to collect liquid waste before transferring the waste either to feed tank 717 for the general purpose evaporator at the Building 211-H Outside Facilities or to tank 21 or 23 at the H-Area tank farm. The destination of the waste depends on the level of radioactivity and organic material in the waste (see Section 9.4.3.4).

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 10

INITIAL TESTING, IN-SERVICE SURVEILLANCE, AND MAINTENANCE

AUGUST 1997

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Official: _____

J.N. DEWES, Manager, Regulatory Programs

Date: _____

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

SAVANNAH RIVER SITE

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ACRONYMS

DOE	Department of Energy
IPI	Installed Process Instrumentation
M&TE	Measuring and Test Equipment
RBOF	Receiving Basin for Offsite Fuels
RRF	Resin Regeneration Facility
SAR	Safety Analysis Report
SFSD	Spent Fuel Storage Division
SRS	Savannah River Site
SSCs	Structures, Systems, and Components
TSR	Technical Safety Requirement
WSRC	Westinghouse Savannah River Company

10.0 INITIAL TESTING, IN-SERVICE SURVEILLANCE, AND MAINTENANCE

10.1 INTRODUCTION

10.1.1 OBJECTIVE

The purpose of this chapter is to provide information that satisfies paragraph 8.b.(3)(o) of Department of Energy (DOE) Order 5480.23, as amplified in Attachment 1, paragraph 4.f.(3)(d)15, of the Order, pertaining to initial testing, in-service surveillance, and maintenance policies and programs (Ref. 1). This chapter includes information that partially satisfies paragraphs 8.b.(3)(b), (f), and (u) of the Order, which are addressed in each chapter of the Safety Analysis Report (SAR) as necessary to provide explicit correlation to topics. The objective of this chapter is to present information demonstrating that the following conditions exist:

- The operations organization of the Receiving Basin for Offsite Fuels (RBOF) and the Resin Regeneration Facility (RRF) is part of a network of policy and programs in terms of organizational structure and interfaces, management policy, initial testing, in-service surveillance, and maintenance of the facility.
- The network of policy and programs is sufficient to ensure that the performed safety-related work meets the initial testing, in-service surveillance, and maintenance requirements of the RBOF and RRF.
- The network of policy and programs includes predictive, preventive, and corrective facility maintenance programs.

The information provided is not intended to be the vehicle for review and approval of the initial testing, in-service surveillance, or maintenance programs, but it is intended to describe the salient features of the policies and programs as they relate to the safety basis of the RBOF and RRF.

10.1.2 SCOPE

Chapter 10 of the Generic SAR (Ref. 2) describes the Westinghouse Savannah River Company (WSRC) initial testing, in-service surveillance, and maintenance programs for the Savannah River Site (SRS), including requirements for these programs, the site organizations that support these programs, and the responsibilities and relationships between these organizations. This chapter provides information on the policies and programs to demonstrate that RBOF and RRF operations include initial testing, in-service surveillance, and maintenance and identifies the site-, division-, department-, and facility-level procedures that implement these programs.

The scope of this chapter is to provide detailed descriptions of the following programs related to the safety-class and safety-significant Structures, Systems, and Components (SSCs) in the RBOF/RRF (i.e., a Hazard Category 2 facility):

- Initial (startup and restart) testing program
- In-service surveillance program
- Planned, periodic, preventive, and corrective maintenance programs

Chapter 4 identifies the safety-class and safety-significant SSCs in the RBOF and RRF.

10.2 REQUIREMENTS

Section 10.2 of the Generic SAR (Ref. 2) identifies the DOE Orders and WSRC documents that specify the codes, standards, and requirements for the initial testing, in-service surveillance, and maintenance programs for SRS and that are applicable to the RBOF and RRF. This chapter identifies the site-, division-, department-, and facility-level procedures that implement these requirements.

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10.3 INITIAL TESTING PROGRAM

This section discusses the testing programs that ensure operability of the RBOF and RRF after a modification and before return to service. Section 10.3 of the Generic SAR (Ref. 2) addresses the testing and operational readiness assessment program for startup activities after a major facility modification (i.e., one that affects the DOE-approved safety basis of the facility). The Generic SAR section also discusses the requirements and activities associated with the restart of SRS facilities, processes, equipment, and systems after a minor modification (i.e., one that does not affect the DOE-approved safety basis of the facility). The RBOF and RRF follow the site-level program for these startup and restart activities in accordance with WSRC Procedure Manual 12Q (Ref. 3).

10.3.1 TESTING OF MODIFICATIONS

Section 10.3.1 of the Generic SAR (Ref. 2) discusses the WSRC requirements for facility configuration management, restart of a facility after a major modification (i.e., one that affects the DOE-approved safety basis of the facility), and implementation of a test plan to ensure readiness of the facility or SSC to operate after being returned to service. These requirements are also applicable in the event of a major expansion of the facility or the installation of new equipment in the facility.

The RBOF and RRF test modifications using the requirements of WSRC Procedure Manuals 7E, 1E, and 1Q (Ref. 4, 5, 6). WSRC Procedure Manual 12Q (Ref. 3) provides directions for determining the type of assessment required for startup.

10.3.1.1 Testing of Facility Modifications

The Spent Fuel Storage Division (SFSD) has adopted WSRC Procedure Manual 5E (Ref. 7). Instead of developing implementing procedures for each section of the manual, procedures from WSRC Procedure Manuals 5E and 5E1 (Ref. 7, 8) are referenced by WSRC Procedure Manual RD-1 (Ref. 9).

10.3.1.2 Testing of Equipment Modifications

Section 10.3.1.2 of the Generic SAR (Ref. 2) discusses the WSRC requirements for post-modification testing of equipment to demonstrate the following:

- Equipment performance of intended functions
- Documentation of test results to ensure equipment operability
- Management review of the test documents to verify correction of the original problem

- Performance of controlled activities to ensure that the equipment is ready to be returned to service

WSRC Procedure Manual 12Q and DOE Order 5480.31 (Ref. 3, 10) consider equipment to be part of the facility's SSCs. Therefore, the requirements discussed in Section 10.3.1.1 for a facility modification apply to the new installation and modification of equipment.

The restart or startup testing requirements after a major equipment modification (i.e., one that affects the DOE-approved safety basis of the facility) or after the installation of new equipment are normally performed by the project restart/startup group in accordance with WSRC Procedure Manual 5E1 (Ref. 8) or a project-specific startup test plan prepared by the facility's project restart/startup group on the basis of WSRC Procedure Manual 5E (Ref. 7).

Restart tests after minor modifications are handled by facility maintenance and operations personnel in accordance with approved maintenance and operations procedures as discussed in Section 10.5.

10.3.2 ADEQUACY OF TESTING ACTIVITIES

Section 10.3.2 of the Generic SAR (Ref. 2) discusses the adequacy of testing activities for the facility and equipment prior to startup or prior to restart after modification. The RBOF and RRF follow the program and procedures contained in WSRC Procedure Manuals 12Q and 5E (Ref. 3, 7) to ensure the safety of the facility and personnel.

10.4 IN-SERVICE SURVEILLANCE PROGRAM

Section 10.4 of the Generic SAR (Ref. 2) discusses the WSRC requirements for in-service surveillance, testing, and facility inspections. Section 10.5 discusses the implementation of DOE requirements for conduct of maintenance.

Surveillance and testing requirements for the RBOF and RRF are identified, performed, and controlled as specified in the Technical Safety Requirement (TSR) (Ref. 11) and 1600 series procedures for the facility. These documents provide guidelines to establish, schedule, implement, and track the RBOF and RRF surveillance test program and identify associated responsibilities, frequencies, notifications, reports, and records. The RBOF/RRF 1600 series procedures establish and maintain a computerized database for scheduling and tracking surveillance tests.

10.4.1 PROVISIONS FOR TESTING AND CALIBRATIONS

Section 10.4.1 of the Generic SAR (Ref. 2) discusses the following:

- WSRC provisions for adequate surveillance tests and test equipment calibration to ensure equipment operation within safe limits
- Recommended frequency of surveillance tests
- Guidelines on planning
- Performance and documentation of surveillance tests

The RBOF TSR (Ref. 11) specifies surveillance requirements, including surveillance frequencies, for safety-class and safety-significant SSCs. Chapter 4 identifies the safety-class and safety-significant SSCs for the facility. Chapter 5 discusses the TSR and associated surveillance requirements.

10.4.2 CONTROL AND CALIBRATION OF MEASURING AND TEST EQUIPMENT

Section 10.5.5 addresses the control and calibration of Measuring and Test Equipment (M&TE) and Installed Process Instrumentation (IPI) used for maintenance. The information in Section 10.5.5 also applies to M&TE and IPI used for in-service surveillance activities.

10.4.3 TRENDING OF SURVEILLANCE TEST RESULTS

The WSRC lessons learned policy is implemented through requirements in WSRC Procedure Manual 1B (Ref. 12). Quality-related information is monitored, evaluated, and reported for the purpose of trending activities through implementation of requirements in WSRC Procedure Manual 1Q (Ref. 6). WSRC Procedure Manual E7 (Ref. 13) specifies requirements for the

identification, testing, collection, and analysis of performance data for SSCs to improve equipment reliability and availability through early detection of equipment degradation.

As indicated in the RBOF/RRF 1600 series procedures, a computerized database is used to schedule and track surveillance tests. Surveillance test results are incorporated into the maintenance history file, discussed in Section 10.5.6, for use in analyzing trends and failures in equipment performance and making adjustments to the maintenance program.

10.4.4 PROGRAMMATIC REVIEW

Section 10.4.4 of the Generic SAR (Ref. 2) discusses the WSRC requirements for in-service inspections and for periodic management assessment and correction, if needed, of the surveillance program, through periodic management reviews of the documented surveillance results and inspections of site equipment and facilities.

Assessments are performed to ensure that the RBOF and RRF are being operated in accordance with the TSR and other DOE guidance. These assessments include the following elements:

- Conformance with the TSR
- Training and qualification of the RBOF and RRF staff
- Program implementation
- Effectiveness of corrective actions
- Quality program adherence
- Other activities of safety significance

The RBOF and RRF assessment program is specified in the administrative controls section of the RBOF TSR (Ref. 11). Chapter 17 of the Generic SAR (Ref. 14) provides additional information regarding self-assessments performed by the facility and independent assessments performed by facility evaluation boards.

10.4.5 TRAINING OF SURVEILLANCE TESTING PERSONNEL

Section 10.5.2 addresses training for maintenance personnel. The information in Section 10.5.2 also applies to training for surveillance testing personnel.

10.5 MAINTENANCE PROGRAM

Section 10.5 of the Generic SAR (Ref. 2) describes the WSRC requirements for the site maintenance program required by DOE Order 4330.4B (Ref. 15), which establishes guidelines for the conduct of maintenance at DOE nuclear facilities. The maintenance implementation plan for SFSD (Ref. 16), which is required by the Order to address each of the DOE guidelines, includes the RBOF and RRF and discusses application of the graded approach by which the SFSD maintenance program is brought to the level of the Order.

As indicated in the Generic SAR, WSRC Procedure Manual 1Y (Ref. 17) provides uniform maintenance policies, requirements, and procedures, as well as division-specific addenda, to implement the requirements of DOE Order 4330.4B. SFSD, which includes the RBOF and RRF, satisfies those elements of the site maintenance program that are completed. For those elements of the site maintenance program that are not yet completed, SFSD conducts maintenance activities in accordance with division-level procedures. For certain program elements, even after the associated Manual 1Y procedures are completed, SFSD may continue to use division-level procedures; however, in those instances, the division-level procedures will be at least as restrictive as the associated Manual 1Y requirements.

10.5.1 MAINTENANCE ORGANIZATION AND ADMINISTRATION

Section 10.5.1 of the Generic SAR (Ref. 2) discusses the WSRC requirements for developing and administering an integrated approach to maintenance that clearly defines the responsibilities of maintenance personnel and establishes working relationships among all maintenance support organizational units. The Generic SAR section also discusses the roles of different parts of the central maintenance organizations, such as the Senior Maintenance Management Council, which serves as the focal point for interfacing with DOE on sitewide maintenance issues, and the Maintenance Training Subcommittee, which delivers and coordinates the training of maintenance personnel.

WSRC Procedure Manual 1Y (Ref. 17) addresses the maintenance organization and administration for all SRS facilities. WSRC Procedure Manual RD-1 (Ref. 9) describes the structure and interdisciplinary functions of the organizations within SFSD. WSRC Procedure Manuals RD3.1 and R3.2 (Ref. 18, 19) establish the role and responsibilities of the SFSD maintenance departments.

10.5.2 TRAINING AND QUALIFICATION OF MAINTENANCE PERSONNEL

Section 10.5.2 of the Generic SAR (Ref. 2) delineates the WSRC requirements for developing and maintaining an effective maintenance training and qualification program. The maintenance manager and supervisors are required to be directly involved in training maintenance personnel; the Maintenance Training Subcommittee oversees this involvement. In addition, the

maintenance program involves the overall support of the maintenance organization with regard to the following:

- Defining the skill levels and responsibilities of various positions
- Defining the training needs
- Providing instructors and training
- Establishing qualification criteria

As indicated in the Generic SAR, WSRC Procedure Manual 1Y (Ref. 17) addresses the training and qualification of maintenance personnel for all SRS facilities.

DOE Order 5480.20A (Ref. 20) provides the requirements for establishing performance-based training programs and personnel qualification requirements for DOE nuclear facilities. This DOE Order is implemented at the site level by WSRC Procedure Manual 4B (Ref. 21). Training requirements for maintenance personnel, as well as surveillance testing personnel, are implemented in SFSD through the following division-level manuals:

- WSRC Procedure Manual RD-1 (Ref. 9)
- WSRC Procedure Manual RD3.1 (Ref. 18)
- WSRC Procedure Manual R3.2 (Ref. 19)

10.5.3 MAINTENANCE FACILITIES, EQUIPMENT, AND TOOLS

Section 10.5.3 of the Generic SAR (Ref. 2) discusses the WSRC requirements for a program to evaluate the adequacy of maintenance facilities, equipment, and tools through facility condition inspections and condition assessment surveys. WSRC Procedure Manual 1Y (Ref. 17) addresses the implementation of these requirements for all SRS facilities.

WSRC Procedure Manual RD3.1 (Ref. 18) provides for SFSD maintenance administration. WSRC Procedure Manual R3.2 (Ref. 19) provides the methods and controls which ensure that important maintenance equipment and tools are readily available and are adequately maintained to perform maintenance functions when needed. Section 10.5.5 addresses the control and calibration of M&TE and IPI used for maintenance and in-service surveillance activities.

10.5.4 POST-MAINTENANCE TESTING

Section 10.5.4 of the Generic SAR (Ref. 2) discusses the WSRC requirements for the performance of post-maintenance testing to verify that SSCs are capable of performing their intended function when returned to service after maintenance and to ensure that the original deficiency is corrected and that no others are created. WSRC Procedure Manual 1Y (Ref. 17) addresses the implementation of these requirements for all SRS facilities.

WSRC Procedure Manual RD3.1 (Ref. 18) defines the post-maintenance testing program as applied to all SFSD facilities for verifying that SSCs are capable of performing their intended function when returned to service following maintenance and for ensuring that the original deficiency is corrected. These requirements are further implemented by the RBOF TSR (Ref. 11) and 1600 series procedures.

10.5.5 CONTROL AND CALIBRATION OF MEASURING AND TEST EQUIPMENT

Section 10.5.5 of the Generic SAR (Ref. 2) discusses the WSRC requirements for instituting a control and calibration program for M&TE to ensure accurate performance of facility instrumentation and equipment used for testing, calibration, and repairs. The Generic SAR section also discusses the requirement for the following:

- Documentation of M&TE calibration results
- Maintenance of a master inventory list of all calibrated M&TE
- Establishment of an M&TE history file
- Unique identification of the calibrated and out-of-calibration M&TE

WSRC Procedure Manual 1Y (Ref. 17) addresses the implementation of these requirements for all SRS facilities.

Currently, M&TE and IPI used by SFSD facilities are controlled, calibrated, and used in accordance with the requirements of WSRC Procedure Manual 1Q (Ref. 6). The Manual 1Q requirements ensure that devices used to verify the measurements of other equipment or to calibrate and accept the measuring devices have documentation of calibrated accuracy traceable to known standards.

10.5.6 MAINTENANCE HISTORY

Section 10.5.6 of the Generic SAR (Ref. 2) discusses the WSRC requirements for establishing and maintaining an equipment repair history and trending program. The Generic SAR section also discusses the inclusion of surveillance test data and past performance problems as part of the equipment repair history and trending program. WSRC Procedure Manual 1Y (Ref. 17) addresses the implementation of these requirements for all SRS facilities.

Currently, maintenance records for SFSD facilities are established and maintained in accordance with requirements in WSRC Procedure Manuals RD-1, RD3.1, and R3.2 (Ref. 9, 18, 19). Trending as part of the site lessons learned program is implemented through requirements in WSRC Procedure Manual 1B (Ref. 12).

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 11

OPERATIONAL SAFETY

AUGUST 1997

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ADC &
Reviewing
Official: JN Dewes

J.N. DEWES, Manager, Regulatory Programs

Date: 9/23/97

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

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ACRONYMS

ARP	Alarm Response Procedure
DOE	Department of Energy
FHA	Fire Hazards Analysis
NOP	Normal Operating Procedure
RBOF	Receiving Basin for Offsite Fuels
RRF	Resin Regeneration Facility
SAR	Safety Analysis Report
SFSD	Spent Fuel Storage Division
SRS	Savannah River Site
WSRC	Westinghouse Savannah River Company

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11.0 OPERATIONAL SAFETY

11.1 INTRODUCTION

11.1.1 OBJECTIVE

The purpose of this chapter is to provide information that satisfies the requirements of Department of Energy (DOE) Order 5480.23, paragraph 8.b.(3)(q), as amplified in Attachment 1, paragraph 4.f.(3)(d)17, of the Order (Topic 17) (Ref. 1). The requirements of this Order pertain to operational safety. This chapter also includes information that partially satisfies paragraphs 8.b.(3)(b), (f), and (u) of the Order, which are addressed in each chapter of the Safety Analysis Report (SAR) as necessary to provide explicit correlation to topics (Ref. 2).

11.1.2 SCOPE

Section 11.1.2 of the Generic SAR (Ref. 3) describes the scope of this chapter and includes information on operational safety that is not described elsewhere in this SAR. The scope of this chapter includes the aspects and summary of conduct of operations for the Receiving Basin for Offsite Fuels (RBOF) and the Resin Regeneration Facility (RRF) and a description of the Fire Protection Program for the RBOF and RRF.

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11.2 REQUIREMENTS

Section 11.2 of the Generic SAR (Ref. 3) states the DOE Orders and Westinghouse Savannah River Company (WSRC) documents that specify codes, standards, and regulations governing the operational safety policies and program elements for the RBOF and RRF.

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11.3 CONDUCT OF OPERATIONS

Section 11.3 of the Generic SAR (Ref. 3) discusses the establishment, requirements, and implementation of the Conduct of Operations Program at the RBOF and RRF.

11.3.1 SHIFT ROUTINES AND OPERATING PRACTICES

11.3.1.1 Facility Operating Practices

The RBOF and RRF, as part of the Spent Fuel Storage Division (SFSD), follow the operating practices defined in WSRC Procedure Manuals 2S and RD-1 (Ref. 4, 5). Day-to-day operations are conducted in accordance with RBOF/RRF Standing Orders and Normal Operating Procedures (NOPs).

Current Standing Orders for the facility are listed in the RBOF/RRF Standing Order Index, which indicates the following Standing Order categories:

- Procedures
- Communications
- Training
- Shift operations
- Facility operations
- RBOF/RRF specific requirements

Current NOPs for the facility are listed in the RBOF/RRF NOP Index, which indicates the following procedure series:

• RRF procedures	100-999
• General building procedures	1000-1999
• General maintenance procedures	2000-2999
• Cask handling procedures	3000-3999
• Fuel handling procedures	4000-4999
• Miscellaneous procedures	5000-9999

The RBOF and RRF also use special procedures when necessary.

11.3.1.2 Operation During Abnormal or Emergency Conditions

WSRC Procedure Manual 2S (Ref. 4) requires that operators respond to alarms in a timely manner and prohibits the disabling of alarms and annunciators without formal authorization. In addition, Manual 2S calls for the creation of special procedures referred to as Alarm Response Procedures (ARPs) (e.g., References 6 through 16). ARPs are kept simple (usually one page in length) to ensure that prompt action is taken to bring the facility to a safe condition in the event of an alarm. Post-event detailed instructions are captured in full-length NOPs.

11.3.1.3 Authority to Operate Equipment

The RBOF/RRF Shift Supervisor is in charge of all day-to-day operations and is aware of all activities affecting the facility. Section 2.4 identifies the major equipment and components of the RBOF and RRF.

For all fuel handling operations, a dedicated Radiological Control Operations inspector is required to be present. The on-shift crew for fuel movement consists of a certified RBOF fuel handling supervisor and an RBOF fuel handling operator who is either qualified or certified, as appropriate. If the RBOF fuel handling supervisor is not certified, both a certified components handling supervisor and a qualified RBOF fuel handling supervisor shall be present in addition to a qualified RBOF fuel handling operator.

11.3.1.4 Operator Rounds and Tours

The RBOF and RRF conduct rounds using NOP 244-H-1241 (Ref. 17), which satisfies the related requirements of WSRC Procedure Manual 2S (Ref. 4) as detailed in Section 11.3.1 of the Generic SAR (Ref. 3).

11.3.1.5 Personnel Protection

The RBOF and RRF use personnel protective practices and techniques discussed in Section 11.3.1 of the Generic SAR (Ref. 3). Chapters 7 and 8 further discuss personnel protection.

11.3.1.6 Shift Operating Bases

The single base of operations for the RBOF and RRF is the Building 244-H Control Room, which contains procedures, references, and communications equipment to support facility operations.

11.3.1.7 Resetting of Alarms or Protective Devices

Section 11.3.1.7 of the Generic SAR (Ref. 3) discusses the requirements for resetting alarms and protective devices.

11.3.1.8 Potentially Distractive Written Material and Devices

Section 11.3.1.8 of the Generic SAR (Ref. 3) discusses the requirements regarding potentially distractive written material and devices.

11.3.1.9 Key Control

To facilitate control over keys used in day-to-day operations, the RBOF and RRF maintain a key accountability log as specified by WSRC Procedure Manual 2S (Ref. 4). Section 11.3.1 of the Generic SAR (Ref. 3) provides further details of the Key Control Program.

11.3.1.10 Overtime

The RBOF and RRF adhere to the specific guidelines for the use of overtime specified in the administrative controls section of the facility Technical Safety Requirements (Ref. 18).

11.3.2 CONTROL AREA ACTIVITIES

The RBOF and RRF follow WSRC Procedure Manual 2S (Ref. 4) for control area activities. The Savannah River Site (SRS) Control Area Activities Program is detailed in Section 11.3.2 of the Generic SAR (Ref. 3).

11.3.3 COMMUNICATIONS

The RBOF and RRF follow WSRC Procedure Manual 2S (Ref. 4) for communications. The SRS Communications Program is detailed in Section 11.3.3 of the Generic SAR (Ref. 3).

11.3.4 CONTROL OF ON-SHIFT TRAINING

The RBOF and RRF follow WSRC Procedure Manuals 2S and 4B (Ref. 4, 19) for the control of on-shift training. The SRS program for controlling on-shift training practices is detailed in Section 11.3.4 of the Generic SAR (Ref. 3).

11.3.5 CONTROL OF EQUIPMENT AND SYSTEMS STATUS

The RBOF and RRF follow WSRC Procedure Manual 2S (Ref. 4) for the control of equipment and systems status. The SRS program for controlling equipment and systems status is detailed in Section 11.3.5 of the Generic SAR (Ref. 3).

11.3.6 LOCKOUTS AND TAGOUTS

The RBOF and RRF use WSRC Procedure Manual 8Q (Ref. 20) to perform lockouts and tagouts as specified by WSRC Procedure Manual 2S (Ref. 4). The SRS Lockouts and Tagouts Program is detailed in Section 11.3.6 of the Generic SAR (Ref. 3).

11.3.7 INDEPENDENT VERIFICATION PRACTICES

The RBOF and RRF follow WSRC Procedure Manual 2S (Ref. 4) for independent verification practices. The SRS Independent Verification Practices Program is detailed in Section 11.3.7 of the Generic SAR (Ref. 3).

11.3.8 LOGKEEPING

The Conduct of Operations Program detailed in WSRC Procedure Manual 2S (Ref. 4) specifies the requirements for establishing and maintaining operating logs. RBOF/RRF logkeeping practices are detailed in Standing Order 244-H-2S-2.4 (Ref. 21). Section 11.3.8 of the Generic SAR (Ref. 3) discusses the requirements that direct RBOF/RRF logkeeping.

11.3.9 OPERATIONS TURNOVER

The RBOF and RRF follow WSRC Procedure Manual 2S (Ref. 4) for operations turnover practices. The SRS Operations Turnover Practices Program is detailed in Section 11.3.9 of the Generic SAR (Ref. 3).

11.3.10 OPERATIONS ASPECTS OF FACILITY CHEMISTRY AND UNIQUE PROCESSES

The RBOF and RRF follow WSRC Procedure Manual 2S (Ref. 4) for operational monitoring of facility chemistry and unique processes. The SRS program for operational monitoring of facility chemistry and unique processes is detailed in Section 11.3.10 of the Generic SAR (Ref. 3).

Section 2.5 describes the RBOF/RRF chemistry and processes. The RBOF and RRF use NOP 244-H-1013 (Ref. 22) to support monitoring activities.

11.3.11 REQUIRED READING

Section 11.3.11 of the Generic SAR (Ref. 3) discusses the requirements of the Required Reading Program.

11.3.12 TIMELY ORDERS TO OPERATORS

Shift Orders are issued to communicate short-term information to operations personnel, and Standing Orders are issued to communicate long-term information to operations personnel. Section 11.3.12 of the Generic SAR (Ref. 3) discusses the guidelines that direct the issuance of orders to RBOF and RRF personnel.

11.3.13 OPERATOR AID POSTINGS

The RBOF and RRF follow WSRC Procedure Manual 2S (Ref. 4) for operator aid postings. The Operator Aid Posting Program describes the request, authorization, documentation, placement, and review required to ensure that operator aids are current, complete, and necessary. Section 11.3.13 of the Generic SAR (Ref. 3) discusses the guidelines that control the use of operator aids.

11.3.14 SYSTEM AND COMPONENT LABELING

The SFSD System and Component Labeling Program, which is implemented by the RBOF and RRF, is established by WSRC Procedure Manual RD-1 (Ref. 5).

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11.4 . FIRE PROTECTION

11.4.1 FIRE HAZARDS

Thorough quantitative Fire Hazards Analyses (FHAs) have been performed for the RBOF (Ref. 23), Support Facility Expansion (Ref. 24), and the Rain Cover Addition (Ref. 25). These FHAs concluded that the RBOF/RRF complex is well maintained and meets the intent of the objectives of the DOE Orders for fire protection. The personnel are well trained in the principles of fire safety, the facility is well staffed, and the principles of fire safety are used in the day-to-day operations of the facilities.

11.4.2 FIRE PROTECTION PROGRAM AND ORGANIZATION

Requirements of the SRS Fire Protection Program, detailed in WSRC Procedure Manual 2Q (Ref. 26), ensure that in the event of a fire the public is protected from uncontrolled releases of radioactive materials to the environment. The program uses fire detection and extinguishing systems and equipment, administrative controls and procedures, and trained personnel to meet these goals. The following principles have been established to prevent and mitigate the consequences of potential fires:

- Prevention of fires through strict administrative control of ignition sources and combustible and flammable materials and through employee training
- Prompt detection and suppression of fires

The buildings in the RBOF and RRF do not have any automatic fire protection or detection systems. The only protection consists of portable dry chemical and Halon™ fire extinguishers. The buildings are provided with hydrant coverage from the 12-inch underground main to the north of the facilities. In addition, Buildings 244-H and 245-H have some passive protection features.

The SFSD Fire Protection Program, which is implemented by the RBOF and RRF, is established by WSRC Procedure Manual RD4.1 (Ref. 27) and is designed to satisfy requirements in WSRC Procedure Manual 2Q and DOE Order 5480.7A (Ref. 26, 28). The following facility-specific documents support the Fire Protection Program at the RBOF and RRF:

- NOP 244-H-1242 (Ref. 29)
- NOP 244-H-8005 (Ref. 30)
- Procedure 244-H in WSRC Procedure Manual 2Q2-4-H (Ref. 31)

In addition, fire extinguishers and battery-powered emergency lights are inspected as tickler items.

As noted in the RBOF FHA (Ref. 23), several minor deficiencies were identified regarding facility compliance with DOE Order 5480.7A. The Compliance Assessment and Implementation Report (Ref. 32) provides a plan and schedule for implementing a Fire Protection Program for all the facilities at SRS, including the RBOF and RRF. The schedule and plan in this report ensure implementation of a Fire Protection Program that complies with DOE Order 5480.7A.

The SRS Fire Protection Program and organization are detailed in Section 11.4.2 of the Generic SAR (Ref. 3).

11.4.3 COMBUSTIBLE LOADING CONTROL

The SFSD Combustible Loading Control Program, which is implemented by the RBOF and RRF, is established by WSRC Procedure Manual RD4.1 (Ref. 27).

11.4.4 FIREFIGHTING CAPABILITIES

Procedure 244-H in WSRC Procedure Manual 2Q2-4-H (Ref. 31) is the RBOF/RRF fire response preplan.

The RBOF and RRF follow WSRC Procedure Manual 2Q (Ref. 26) for firefighting capabilities. The SRS Firefighting Capabilities Program is detailed in Section 11.4.4 of the Generic SAR (Ref. 3).

11.4.5 FIREFIGHTING READINESS ASSURANCE

The RBOF and RRF follow WSRC Procedure Manual 2Q (Ref. 26) for firefighting readiness assurance. The SRS Firefighting Readiness Assurance Program is detailed in Section 11.4.5 of the Generic SAR (Ref. 3).

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2. Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports. DOE-STD-3009-94, U.S. Department of Energy, Washington, DC, July 1994.
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4. Conduct of Operations Manual (U). WSRC Procedure Manual 2S, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
5. Spent Fuel Storage Division Administrative Manual (U). WSRC Procedure Manual RD-1, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
6. Instrument Air Low Pressure Alarm (U). ARP 244-H-ARP-480.10-A6, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
7. Vessel Vent Exhaust Fan Loss (U). ARP 244-H-ARP-480.10-B3, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
8. Main Exhaust Fan Loss (U). ARP 244-H-ARP-480.10-B4, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
9. 245-H General Alarm (U). ARP 244-H-ARP-480.10-B5, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
10. Expansion Tank HI/LO Alarm (U). ARP 244-H-ARP-480.10-C3, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
11. Chiller 901.1C Fault Alarm (U). ARP 244-H-ARP-480.10-C4, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
12. EP38 High Level (U). ARP 244-H-ARP-480.10-C6, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
13. Waste Tank Very High Level (U). ARP 244-H-ARP-480.10-A3, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
14. Basin Level Alarms (U). ARP 244-H-ARP-480.2-480.9-A2-A10, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.

15. NIM Trouble Storage and Cask Basin (U). ARP 244-H-ARP-NIM-B3, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
16. NIM Trouble Inspection or Disassembly Basin (U). ARP 244-H-ARP-NIM-B4, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
17. RBOF Operator Round Sheets (U). NOP 244-H-1241, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
18. Technical Safety Requirements Receiving Basin for Offsite Fuels Facility and Resin Regeneration Facility (U). WSRC-TS-95-11, Rev. 1, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC, June 1998.
19. Training and Qualification Program Manual (U). WSRC Procedure Manual 4B, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
20. Employee Safety Manual (U). WSRC Procedure Manual 8Q, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
21. Operating Logs (U). Standing Order 244-H-2S-2.4, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
22. RBOF Routine Water Samples (U). NOP 244-H-1013, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
23. Fire Hazards Analysis for RBOF (U). F-FHA-H-00014, Rev. 0, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC, June 1994.
24. Fire Hazards Analysis for Project S-5891, 244-H, Support Facility Expansion (U). F-FHA-H-00027, Rev. 0, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC, November 1995.
25. Fire Hazards Analysis for Project S-017, 245-H, Rain Cover Addition (U). F-FHA-H-00025, Rev. 0, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC, August 1995.
26. Fire Protection Program Manual. WSRC Procedure Manual 2Q, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
27. Reactor Division Fire Protection Program Manual (U). WSRC Procedure Manual RD4.1, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
28. Fire Protection. DOE Order 5480.7A, U.S. Department of Energy, Washington, DC, February 1993.

29. RBOF/RRF Quarterly Chemical Inventory (U). NOP 244-H-1242, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
30. Inspections of the Chemical Storage Areas of RBOF (U). NOP 244-H-8005, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
31. H-Area Fire Control Preplans. WSRC Procedure Manual 2Q2-4-H, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
32. DOE Directive Compliance Assessment and Implementation Report. SRS-DOE-5480.7A-CSA-93-035, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC, October 1993.

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ACRONYMS

DOE	Department of Energy
RBOF	Receiving Basin for Offsite Fuels
RRF	Resin Regeneration Facility
SAR	Safety Analysis Report
SRS	Savannah River Site
WSRC	Westinghouse Savannah River Company

12.0 PROCEDURES AND TRAINING

12.1 INTRODUCTION

12.1.1 OBJECTIVE

This chapter provides information relating to procedures and training programs for the Receiving Basin for Offsite Fuels (RBOF) and the Resin Regeneration Facility (RRF). Chapter 12 of the Generic Safety Analysis Report (SAR) (Ref. 1) discusses the overall processes by which the technical content of procedures and training programs is developed, verified, and validated for the Savannah River Site (SRS). It also discusses the mechanisms used to identify and correct technical or human factors deficiencies in procedures and training programs and provides information regarding compliance with Department of Energy (DOE) requirements.

12.1.2 SCOPE

Chapter 12 of the Generic SAR (Ref. 1) describes the processes by which the technical content of the procedures and training programs is developed, verified, and validated for SRS. The procedures and training processes ensure that facilities are operated and maintained by personnel who are qualified and competent to carry out their job responsibilities. In addition, the Generic SAR discusses the processes used to maintain procedures, training materials, and training records and to modify procedures and training materials through the use of feedback and continuous improvement.

This chapter supplements the Generic SAR in describing the program specific to the RBOF and RRF. Only deviations, additions, or enhancements specific to the RBOF and RRF outside of the programs discussed in the Generic SAR are discussed in detail in this chapter. The procedures and training related to testing, in-service surveillance, and maintenance are discussed in Chapter 10.

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12.2 REQUIREMENTS

The procedures program at the RBOF and RRF is governed primarily by Chapter 16 of DOE Order 5480.19 (Ref. 2). Chapter 16, titled "Operations Procedures," contains guidelines regarding the following:

- Procedure development
- Content
- Changes and revisions
- Review
- Approval
- Availability
- Use

To implement these guidelines, Westinghouse Savannah River Company (WSRC) Procedure Manual 1Q (Ref. 3) states the policies that govern the operations of all facilities at SRS. WSRC Procedure Manual 1B (Ref. 4) specifies the requirements and responsibilities relating to program-specific administrative procedures. WSRC Procedure Manual 2S (Ref. 5) specifies the requirements for the control and development of technical and response procedures and acceptable alternate implementation methods. WSRC Procedure Manual RD-1 (Ref. 6) contains further guidance to implement the DOE Order and site procedure requirements at the RBOF and RRF.

The training programs at the RBOF and RRF are governed by DOE Order 5480.20A (Ref. 7). Section 12.2 of the Generic SAR (Ref. 1) discusses the requirements of DOE Order 5480.20A and WSRC Procedure Manual 4B (Ref. 8), which controls the training program at SRS.

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12.3 PROCEDURES

It is the policy of WSRC that the primary consideration in the operation of SRS facilities is the safety of the general public, employees, facility equipment, and the environment. This policy is accomplished by ensuring that operations and support organizations perform all activities affecting quality in accordance with approved procedures that specify the requirements for performing work in a safe and efficient manner. This policy is stated in WSRC Procedure Manual 1Q (Ref. 3) and is implemented at the site level by WSRC Procedure Manual 2S (Ref. 5) for technical and response procedures and by WSRC Procedure Manual 1B (Ref. 4) for program-specific administrative procedures.

12.3.1 DEVELOPMENT OF PROCEDURES

Section 12.3 of the Generic SAR (Ref. 1) discusses the WSRC manuals that provide details for generating technical and response procedures. WSRC Procedure Manual 2S and WSRC-SCD-2 (Ref. 5, 9) include guidance on the required content, necessary research, and format of procedures, as well as guidance for procedure preparation, validation, and verification. These guidelines apply to both technical procedures (including normal operating procedures, maintenance procedures, test procedures, surveillance procedures, and other procedures that provide step-by-step instructions for the performance of an activity or evaluation) and response procedures (including abnormal operating procedures, emergency operating procedures, and alarm response procedures). Chapter 10 provides additional guidelines for the development and control of test procedures, surveillance procedures, and maintenance procedures. WSRC Procedure Manual 1B (Ref. 4) controls the generation of program-specific administrative procedures. Chapter 15 discusses emergency plan-related procedures.

To implement these requirements for the RBOF and RRF, WSRC Procedure Manual RD-1 (Ref. 6) is used to develop, approve, and control technical, response, and administrative procedures.

12.3.2 MAINTENANCE OF PROCEDURES

Section 12.3 of the Generic SAR (Ref. 1) gives the requirements for the maintenance and control of procedures, including changes and revisions, periodic reviews, control and dissemination, and training and coordination. WSRC Procedure Manual 2S (Ref. 5) provides guidance regarding the maintenance and control of procedures to ensure proper dissemination and use of facility procedures. WSRC Procedure Manual RD-1 (Ref. 6) implements the requirements of Manual 2S for the maintenance and control of procedures.

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12.4 TRAINING

The mission of the SRS Training Program, as discussed in Section 12.4 of the Generic SAR (Ref. 1), is to develop programs that ensure site operations are conducted in accordance with the highest standards of safety, health, and environmental protection. This mission is accomplished by ensuring that personnel performing operations at SRS meet the qualification and certification requirements and are trained in the performance and requirements of their duties.

DOE Order 5480.20A (Ref. 7) provides the requirements for establishing performance-based training programs and the personnel qualification requirements for DOE nonreactor nuclear facilities. This DOE Order is implemented at the site level by WSRC Procedure Manual 4B (Ref. 8).

12.4.1 DEVELOPMENT OF TRAINING

Section 12.4 of the Generic SAR (Ref. 1) summarizes the processes by which the technical content of training programs is developed, reviewed, and approved. These guidelines apply to the training programs that define the communication and coordination activities necessary to carry out technical programs at the RBOF and RRF.

The RBOF and RRF use the site- and division-level training programs for the following:

- Radiation protection
- Hazardous material protection
- Hazards communication
- Fire protection systems
- Fundamentals
- Industrial hygiene
- Manager/supervisor training
- Quality assurance

Facility-specific training is developed and presented in those areas where supplemental training would be beneficial to the safe operation of the facility. Because of the transfer of responsibility for the RBOF and RRF from the Nuclear Materials Stabilization Program Division to the Spent Fuel Storage Division, training has been developed in hazards communications and emergency preparedness specifically for the RBOF and RRF.

The RBOF and RRF use site-, division-, and facility-level initial and continuing training courses, as appropriate. The courses address, but are not limited to, training requirements in the following areas:

- Radiation protection
- Hazardous material protection
- Respiratory protection
- Nuclear criticality
- Hazards communication
- Hazardous energy control
- Waste certification
- Security
- Quality assurance
- Tamper-indicating devices
- Two-person rule
- Fitness for duty
- Principles of operation
- RBOF/RRF processes and systems
- RBOF/RRF authorization basis documents (e.g., Technical Safety Requirements)

12.4.2 MAINTENANCE OF TRAINING PROGRAMS

Section 12.4 of the Generic SAR (Ref. 1) describes the method used to ensure that training programs reflect actual plant conditions and current procedures and that necessary coordination is performed before introducing new training programs or introducing changes in procedures covered by training programs. The Generic SAR covers the development, maintenance, and coordination of training programs and procedures and the maintenance of training records as prescribed by WSRC Procedure Manual 4B (Ref. 8).

12.4.3 MODIFICATION OF TRAINING MATERIALS

Section 12.4 of the Generic SAR (Ref. 1) describes the methods used to identify and correct technical or human factors deficiencies in facility training programs, including the Operating Experience Review Program, personnel examinations, and the Lessons Learned Program. The Generic SAR gives details on the incorporation of changes to training materials, administration of training development requests, and evaluation of training effectiveness.

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 13

HUMAN FACTORS

AUGUST 1997

UNCLASSIFIED DOES NOT CONTAIN UNCLASSIFIED CONTROLLED NUCLEAR INFORMATION	
ADC & Reviewing Official:	<u>J.N. Dewes</u>
J.N. DEWES, Manager, Regulatory Programs	
Date:	<u>9/23/97</u>

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

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J.N. DEWES, Manager, Regulatory Programs

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ACRONYMS

DOE	Department of Energy
DOT	Department of Transportation
HVAC	Heating, Ventilation, and Air Conditioning
HMI	Human-Machine Interface
NFPA	National Fire Protection Association
NIM	Nuclear Incident Monitor
NPH	Natural Phenomena Hazard
RBOF	Receiving Basin for Offsite Fuels
RRF	Resin Regeneration Facility
SAR	Safety Analysis Report
SRS	Savannah River Site
SSCs	Structures, Systems, and Components
TSR	Technical Safety Requirement
WSRC	Westinghouse Savannah River Company

13.0 HUMAN FACTORS

13.1 INTRODUCTION

This chapter describes the influence of human factors on the safety analysis for the Receiving Basin for Offsite Fuels (RBOF) and the Resin Regeneration Facility (RRF). RBOF operations involve manual material handling as would occur in a warehouse with the added complexity that most material must be handled under water. Operators move material, in this case usually irradiated nuclear fuel, from trucks or railroad transport to storage locations. The nuclear fuel may be in the form of individual fuel assemblies or groups of fuel assemblies, or it may be contained in transport casks, depending on the particular step in the storage process. In the process of storing the nuclear fuel, operators perform administrative inventory duties and may also perform repackaging activities. Human actions are relied on for certain controls and operations that affect the safety of the facility. Human actions affecting safety are those associated with monitoring facility parameters that are initial conditions of the Safety Analysis Report (SAR); controlling irradiated fissile nuclear material type, amount, and location; and responding to criticality alarms for prompt evacuation.

The RRF is a chemical resin processing facility. The resin to be processed is received by deionizer truck or transferred through pipes from the RBOF. After processing, the resin is returned to its source or transferred to a holding tank before disposal. Spills of hazardous chemicals are the principal safety concern for the RRF. Human actions affecting safety are primarily associated with following administrative controls on the type and location of hazardous materials and with ensuring proper status of spill control structures.

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13.2 REQUIREMENTS

The RBOF and RRF were designed before detailed and systematic human factors considerations were required in the design process. No documented human factors studies of the facility have been conducted with the exception of the review completed for this chapter. At the time this chapter was written, the facility was in full operation and has been for many years.

13.2.1 HUMAN-MACHINE INTERFACE

Human-Machine Interface (HMI) requirements are provided in Department of Energy (DOE) Order 6430.1A (Ref. 1). HMI requirements focus on the design of specific information and control displays. Examples of topics are information availability (e.g., "Was the parameter, alarm, diagram, etc., provided?"), information perception (e.g., "Is the information large enough, the right color, worded properly, etc.,?"), and communications (e.g., "Are adequate devices provided to support exchange of information?"). This facility was designed before DOE Order 6430.1A became effective. No major modifications requiring the application of DOE Order 6430.1A have occurred in this facility. For these reasons, no human factors requirements, (Section 1300-12 of the Order) were applied to the RBOF and RRF. DOE Order 5480.19 (Ref. 2) sets requirements for controlling equipment identification, status boards, operator aids, and procedure validation and verification.

13.2.2 PHYSICAL WORK ENVIRONMENT

Physical work environment requirements are provided in DOE Order 6430.1A (Ref. 1). The physical work environment includes the following:

- Heating, Ventilation, and Air Conditioning (HVAC)
- Work space lighting
- Work space noise levels
- Work space physical access (i.e., anthropometrics, such as "Can a person physically operate the equipment?"; "Is there room to move around the equipment?"; and "Are there clear routes to the equipment?")
- Personal protective equipment

The HVAC for the RBOF and RRF structures was designed to appropriate HVAC standards.

Work space lighting is designed according to standard industrial requirements used in storage facilities. Du Pont lighting standards were used in the RBOF and RRF to specify lighting requirements under normal and emergency conditions. These standards address lighting needs inside structures (including material handling and storage areas) and in outside areas (including material handling, storage, and tank farm locations). These lighting standards meet or exceed

equivalent National Consensus Standards. Essential lighting requirements are specified in National Fire Protection Association (NFPA) 101-88 (Ref. 3), as well as Du Pont lighting standard that also reference NFPA 101-88. Essential lighting requirements apply to loss-of-power situations.

Noise control requirements focus on noise levels that interfere with communications. RBOF and RRF structures employ standard architectural practice for noise control. Human actions with a significant safety impact in the RBOF and RRF do not usually occur in situations where noise impacts communication. Situations where noise may influence job performance are monitored and procedurally controlled as required in DOE Order 5480.19 (Ref. 2). Requirements also exist to control noise levels that could pose a short- or long-term health hazard. However, these noise levels usually exceed those that interfere with communications and are not applicable in human factors considerations.

Work space physical access is required in RBOF and RRF work areas (e.g., control rooms, basin and cask handling work areas, and chemical storage pads). Physical access refers to the arrangement of the work area layout and materials so that they can be properly accessed for movement and inspection. Good industrial practice and Savannah River Site (SRS) experience are used to determine acceptable physical access.

Personal protective equipment can have influences that are equivalent to HVAC, lighting, noise, and physical access. Personnel using protective equipment must be able to breath, control body temperature, see clearly, communicate (speak and hear), and physically access their work environment. Personal protective equipment meeting federal standards is used when available for providing specific protective requirements of service at SRS. When specialty personal protective equipment must be developed, human-workplace interaction is considered in design and testing before general use is permitted.

13.2.3 ORGANIZATIONAL CONSIDERATIONS

Organizational considerations include the following:

- Minimum staffing levels
- Allocation of control functions
- Overtime restrictions
- Communications between shifts
- Procedures
- Training

DOE Orders 5480.19, 5480.20, and 5480.22 (Ref. 2, 4, 5) define requirements in organizational areas.

13.3 HUMAN FACTORS PROCESS

A human factors review of the RBOF and RRF safety Structures, Systems, and Components (SSCs) and safety programs was performed to generate this chapter. As described in Section 13.2, this facility was designed before DOE human factors design requirements in DOE 6430.1A (Ref. 1) were issued. No initial human factors design reviews have been done for the RBOF and RRF or for facility system upgrades subsequent to initial design.

The human factors review was done based on information provided in the RBOF Functional Classification Report (Ref. 6). This document was used to identify safety SSCs and safety programs. Safety SSCs and safety programs include both safety-class and safety-significant SSCs and programs. The human actions and related HMIs associated with the operation of these SSCs and programs were identified and described in Section 13.4. This information was then used to create human factors checklists as described in WSRC-RP-95-460 (Ref. 7). The human factors checklists are then used to evaluate the human actions and HMIs of the RBOF and RRF. The findings of the review are summarized in Section 13.5. Section 13.4 describes the operator interaction with safety SSCs and safety programs that were analyzed in Section 13.5.

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13.4 IDENTIFICATION OF HUMAN-MACHINE INTERFACES

HMIs for safety SSCs in the RBOF and RRF can be divided into the following three categories based on the function the safety SSCs perform in maintaining facility safety:

- Safety SSCs that monitor the status of the facility so that the initial conditions assumed in the hazards and accident analysis (Chapter 3) remain valid
- Safety SSCs that prevent the assembly of a critical mass in the RBOF basin and shipping cask handling areas
- Safety SSCs that alert personnel to the occurrence of a criticality accident and the need for immediate evacuation

This section summarizes the general facility layout and design features to provide an understanding of the physical environment in which personnel must perform and describes the HMIs in each of the three categories listed above.

13.4.1 FACILITY LAYOUT AND DESIGN FEATURES

All process operations for the RBOF are performed in Building 244-H. Building 245-H contains the RRF. The structures are physically connected and are located within the existing boundaries of H Area. Routine personnel access to Buildings 244-H and 245-H is by means of six doorways. Fuel shipping casks are admitted through a roll-up door in the truck and railroad port of Building 244-H. Portable deionizers (truck-mounted cask/deionizers) are connected to the RRF at an external hose box loading/unloading station located next to Building 245-H. Chapter 2 includes more detailed information and diagrams on the facility layout.

RBOF personnel perform cask handling operations in the carport and from the basin deck areas around the cask wash pit and the cask basin. Cask movement is performed using a 100-ton overhead crane. The overhead crane has mechanically actuated electrical limit switches (an active safety design feature) and mechanical stops (a passive safety design feature) that prevent the crane from reaching the end of the rails or being positioned over the #2 storage basin. These design features eliminate a possible source of human error that could initiate a criticality accident.

Cask decontamination, venting, flushing, sampling, and initial cask disassembly are performed in the cask wash pit. Cask lid bolts are also removed in the cask wash pit with the exception of a few bolts. The cask is then transferred by crane to the cask basin where the remaining bolts and lid are removed. Lid removal is performed under water. All movement and processing of irradiated nuclear fuel is performed under water by operators standing on a traveling bridge platform that spans the basin or the basin deck, using hoists and long-handled tools. The fuel basins and canal system, consisting of fuel handling and storage basins and associated transfer canals, serve as a secondary or tertiary containment for fission products. These structures

provide both a safety-class and safety-significant function. No HMI exists for this SSC with the exception of the basin water level monitoring described later in this section.

The controls for the operation of pumps, valves, and other equipment used for the addition, purification, and disposal of basin water are located in the RBOF Control Room. The RBOF Control Room is separated from the basin area by a shielding wall of concrete. The RBOF shielding and the RRF shielding are safety-significant structures. However, these structures do not have HMIs requiring evaluation. The above-grade structure of Building 244-H is a safety-significant structure that protects stored fuel from Natural Phenomena Hazards (NPHs) and limits the resulting source terms. The intact structure prevents a criticality by not collapsing and impacting fuel stored in the basin. The walls of the building also serve to contain radionuclide releases resulting from criticality accident scenarios. The position (open/closed) of personnel doors, railroad/truck doors, or air system dampers does not affect the safety-significant function, so no building structure HMIs requiring evaluation exist.

The RRF receives mixtures of anion and cation resins from the RBOF basin water deionizer or from portable deionizers used in the SRS reactor disassembly basins. The trailer-mounted portable deionizers are manually connected to the RRF at outdoor hose boxes located outside the RRF Control Room. The controls for the operation of RRF facility pumps, valves, and other equipment used for the regeneration or disposal of deionizer resins are located in or near the RRF Control Room. The RRF Control Room is protected from radiation emanating from the resin regeneration cell and portable deionizers by a concrete shielding wall.

13.4.2 FACILITY STATUS MONITORING SSCs

The HMIs discussed in this section are for both safety-class and safety-significant SSCs that monitor the status of the RBOF so that the initial conditions assumed in the hazards and accident analysis (Chapter 3) remain valid. These SSCs alert operations personnel that a parameter is or may soon be outside the prescribed limit. Limits on the parameters monitored by these SSCs are established in the Technical Safety Requirements (TSRs) (Chapter 5). The safety SSC and administrative controls that perform this monitoring are as follows:

- An instrument system with an HMI in the form of annunciators and an associated parameter display. The annunciators alert personnel with a visual signal in the form of an illuminating alarm plate and audio signal in the form of a horn. The parameter display is in the form of a digital or analog gage.
- An administrative control system where personnel obtain information from a document and compare it to limits prescribed in that document or another document.

Both instrument and administrative control systems include procedures and training and may include operator aids or status boards as part of the overall system.

13.4.2.1 Basin Water Level

Deionized water is used in RBOF's water-filled basins. The basin water provides shielding to protect facility workers from radiation during normal operations, and in the event of a criticality, from prompt neutrons and gamma radiation. The basins are normally connected such that the water level is uniform. Adequate basin water level is assumed as an initial condition for criticality accidents. Basin water is credited as an attenuation medium for reducing prompt worker exposure during a criticality event and provides scrubbing of any released fission products in the event of fuel melting during the criticality event. Therefore, a minimum required basin level is established in the TSRs.

The RBOF basin water system consists of the following:

- A cask unloading basin
- Two storage basins
- A disassembly basin
- An inspection basin
- A repackaging basin
- A transfer canal that is divided into two sections

The #1 storage basin has a permanently mounted ruler for basin water level indication. This passive measurement device performs both a safety-class and safety-significant function. The design of the ruler provides for normal and abnormal range indication of basin water level. The specific limits on water level and actions that must be taken if the level is low are described in the TSRs. The basins are interconnected so the #1 storage basin level is valid for the entire basin system. The routine monitoring (operator rounds) provides the primary means of detecting basin water leakage. The margin between the low level indication and the TSR limit is intended to be sufficient to permit human response. Response procedures are provided to correct basin level problems.

The RBOF may be unstaffed when the facility is shut down. During this time, a decreasing water level would be detected by the Nuclear Incident Monitor (NIM) alarm in the Building 221-H Control Room (see Section 13.4.4.1), due to high radiation levels. A problem with the NIM units would be indicated by the Remote 244-H General Alarm (see Section 13.4.4.2), which has an input from the NIM trouble circuit. This alarm may be activated by several other parameters besides NIM trouble. The Building 221-H procedures direct an operator to go to Building 244-H, determine the cause of the alarm, and take procedurally directed action to correct the problem or contact the Building 221-H Control Room for assistance.

13.4.2.2 Basin and Basin Deionizer Radioactivity Level

Radioactivity in the RBOF basin water results from breached fuel cladding. The activity is primarily radioactive cesium. Other radioactive isotopes, as well as activated corrosion products in lesser amounts, may also be found in the basin water. A limit on basin and basin deionizer activity minimizes a potential release to the environment during normal operation, anticipated operational occurrences, and accidents.

Periodic basin water samples are taken and sent to onsite or offsite laboratories for analysis. The HMI for this process consists of the labels and documentation that accompany the samples to the laboratories and the documentation of laboratory analysis results that are returned to the RBOF. Human factors analysis of laboratory equipment is beyond the scope of this SAR. Training, Standing Orders, and procedures are in place for RBOF personnel collecting samples and evaluating laboratory results returned to the RBOF. Labeling of sample locations is controlled programmatically. Personnel response to sample results that are outside of acceptable limits are defined in written procedures. The margin between the procedure limits and safety analysis assumptions is intended to be sufficient so that action can be taken to prevent exceeding the value analyzed. TSR administrative programs address the required protection for the activity levels in the basin water.

RBOF deionizer resin is monitored (by monitoring activity in the basin water) and regenerated or replaced to limit radionuclide buildup. This radionuclide inventory is an important factor in certain accident scenarios. The resin parameters monitored include the types and quantities of radioactive material for resin deposit, the rates of accumulation, resin service time, and deionizer flow rates. Basin water radionuclide laboratory analysis, deionizer flow instrumentation, training, and procedures are provided to track deionizer inventory.

13.4.2.3 Basin Water Chemistry Control Program

The Basin Water Chemistry Control Program is required to protect against corrosion to ensure the long-term integrity of spent fuel stored in the RBOF basins and basin hardware (e.g., racks, baskets, spacers, underwater tools). Multiple failures of stored fuel could cause releases of radioactive material to the basin, severely complicating basin operations. Multiple mechanical failures of nuclear fuel, handling hardware, or storage structures, particularly during transfer operations, could potentially result in a nuclear criticality. Maintaining nuclear fuel and basin hardware integrity, by controlling corrosion, reduces the risk of multiple component failures and operating problems associated with high basin activity.

In addition to limiting corrosion, the Basin Water Chemistry Control Program maintains water clarity. Water clarity is essential in performing most underwater tasks safely. All fuel and cask handling operations require acceptable visibility. The water must be clear enough to permit detailed operations, such as inspection of fuel condition and reading of fuel serial numbers.

The Basin Water Chemistry Control Program involves both monitoring, sampling, and analysis of basin water. Instrumentation for monitoring basin water temperature (increases in temperature can increase the occurrence of chemical reactions) is provided in the RBOF. Temperature is measured at the #1 storage basin. Conductivity is measured weekly. Conductivity monitoring protects against inadvertent addition of corrosive material caused by personnel errors or spills. Conductivity instrumentation is also provided which monitors the deionizer outlet. Procedures and training to conduct this monitoring are provided.

Periodic basin water samples are taken to measure pH, conductivity, copper, chloride, and mercury levels. These samples are sent to onsite or offsite laboratories for analysis. The HMI for this process consists of the labels and documentation that accompany the samples to the laboratories and the documentation of laboratory analysis results that are returned to the RBOF. Human factors analysis of laboratory equipment is beyond the scope of this SAR. Training and procedures are in place for RBOF personnel collecting samples and evaluating laboratory results returned to the RBOF. Labeling of sample locations is controlled programmatically. Personnel response to sample results that are outside of acceptable limits are defined in the RBOF TSR (Ref. 8).

13.4.2.4 Spent Fuel Parameters and Shipping Cask Coolant Controls

Spent fuel parameters constitute initial conditions in the hazards and accident analysis (Chapter 3). These parameters are as follows:

- Fuel characteristics
- Minimum cooling time
- Maximum heat load
- Maximum radionuclide inventory for an individual fuel unit
- Maximum facility radionuclide inventory
- No sodium contained in fuel or as fuel surface contamination

These parameters cannot be measured in the RBOF. Higher than expected temperature or radiation levels, measured when casks are received, may indicate failure to meet some of these parameters. However, these are not reliable indicators that these parameters have or have not been violated.

Shipping casks containing antifreeze cannot be purged to waste tank EP38. No procedures or hardware exist for adding antifreeze to onsite shipping casks. Appendix B to the offsite cask shipping contract states that the use of any liquid coolant other than water as the primary cooling medium, or the use of additives, such as antifreeze, will be considered on an individual basis prior to shipment and may require special handling. Identification of a cask shipped with coolant other than water relies on the information being given in the shipping documentation. No coolant is sampled before purging to waste tank EP38.

Fuel information is transmitted to the RBOF in standardized documents specified by DOE and the U.S. Department of Transportation (DOT). Training, administrative controls, and operating procedures are provided to RBOF operations personnel to interpret these documents and evaluate the status of the spent fuel. WSRC-TR-95-0302 (Ref. 9) provides documentation for RBOF specific limits and discussion to support the safety documentation and facility operations. Specific limits on spent nuclear fuel and onsite shipping casks are contained in the RBOF TSR (Ref. 8). These limits undergo multiple reviews by Engineering, Operations, and Nuclear Criticality Safety organizations.

13.4.2.5 Hazardous Chemical Limits

A Hazardous Material Inventory Control Program sets maximum inventory limits for hazardous materials in the RBOF and RRF. The program provides control over new materials brought into the facility and also maintains chemical inventories within the levels assumed in the SAR. Administrative controls, including procedures, training, labeling, status boards, and operator aids, are the primary human factors aspects of the HMI and are controlled programmatically. Control of hazardous material is discussed in detail in Chapter 8.

13.4.2.6 Chemical Pad No. 2 Tanks and Berm

The RRF chemical pad No. 2 sodium nitrite tanks and berm work together to contain sodium nitrite and to prevent the mixing of this chemical with nitric acid. The tanks and berm are a safety-significant SSC.

The sodium nitrite tanks contain the sodium nitrite during design basis NPHs and limit pad inventory to that which can be contained in the berm in the event of a release. The HMIs of the system are inspections which ensure that the volume of the bermed areas is not degraded through excessive storage of items in the berm. Inspection of the bermed areas of chemical pad No. 2 is specified in procedures that are performed periodically. The pads are verified to be free of objects that could reduce the contained volume of the bermed areas.

13.4.3 CRITICAL MASS PREVENTION SSCs

The HMIs discussed in this section are for safety-significant SSCs that prevent the assembly of a critical mass in the RBOF basin and shipping cask handling areas (see Chapter 6). These SSCs are primarily structures and/or components that maintain fuel spacing or provide neutron absorption in the storage basins or in the shipping casks during transport and loading/unloading operations. A system of administrative controls to limit the number of fissile units (restricting the total mass of fissile material) in process is used when physical structures are not available or practical.

Fundamental to the safe performance of manual underwater operations (performed with long-handled tools) and inspections is the ability of the operators to clearly see objects under water. This ability requires adequate lighting, above and below the water, and water of sufficient clarity to perform the tasks. Binoculars are also provided when tasks require detailed inspection or reading of information from objects that cannot be brought close enough to the basin surface. Water clarity is maintained by procedural limits on turbidity and dissolved solids. Lighting is periodically inspected and preventative maintenance is performed.

13.4.3.1 Fissile Material Spacing and Neutron-Absorbing SSCs

RBOF basin structures and components that maintain spacing include storage racks, grating, row ties, row latches, I-beams, containers, baskets, and basket lids. Cask inserts that maintain fuel spacing during transport and loading/unloading include spacers, blanks, fuel containers, and baskets. Both onsite and offsite casks may use neutron-absorbing components for nuclear safety. The neutron-absorbing materials are usually structural components like the components that maintain spacing. These are typically referred to as poison plates. However, the neutron absorbers may be inserted directly into the fuel in the form of control rods or poison inserts. Procedures, training, engineering drawings, and labeling are provided for the installation, modification, operation, and inspection of spacing and neutron-absorbing components.

Each fuel type received in the RBOF is placed in approved storage locations that maintain the proper spacing. Storage locations are labeled in agreement with procedures and storage records. Most fuels in the storage basins are hidden from view because they are stored in bundles, and some fuels are hidden from view because they are stored beneath rows of other fuel. Therefore, visual inspection cannot directly verify basin content. The RBOF Fuel Storage Array sheet and Supervisor's and Clerk's Fuel Storage Books provide the principal basin content status. The RBOF Fuel Storage Array sheet is a summary of all fuel in RBOF storage and used to perform the physical inventory. The RBOF Fuel Storage Array sheet also includes loaded casks parked outside of Building 244-H. The Supervisor's and Clerk's Fuel Storage Books identify fuel by serial number engraved on the fuel or by waterproof black ink markings on fuel tube bundles or other stored items. Fuel serial numbers are read with binoculars if not large enough to read directly under several feet of water.

13.4.3.2 Fissile Material Unit Control

When critically safe spacing cannot be ensured, the number of fissile units is limited. Fissile material unit control is used during fuel processing including cask unloading/loading, transfer, inspection, repackaging, and placement for storage. Temporary storage of fissile units is permitted at pre-designated locations that are marked with nuclear criticality safety chains and signs. Fissile units combined in bundles or baskets are treated as one unit. Integrity of these units is verified by visual inspection of bundle tabs or basket lids and tactile feedback ("the feel") of the long-handled tools used in the repackaging process. Underwater tools used by RBOF

personnel are subject to ergonomic analysis, described in SRS personnel safety procedures (Ref. 13), to ensure proper control of underwater operations.

13.4.3.3 Cask Handling Operations

Critically safe handling of offsite shipping casks relies on engineering drawings, written procedures, and shipping documents provided by the shipper and the cask owner. Cask handling and loading/unloading training and procedures are developed by RBOF personnel from this information, primarily from the procedures contained within the Safety Analysis Report for Packaging. Onsite casks are handled in accordance with facility procedures. Also, special underwater tools may have to be developed for these operations. These special handling tools are subject to ergonomic analysis (Ref. 13) to ensure proper control of underwater operations. Shipping cask placards, labels, and markings, as required by DOT, may provide important information to RBOF personnel. The reported contents of the received cask must be verified with the definitive load list before the cask is unloaded. If multiple casks are received, minimum allowed spacing mandated by DOT regulations between casks must be maintained by RBOF personnel.

13.4.4 ACCIDENT ALERTING AND MITIGATION SSCs

The HMIs discussed in this section are for safety-significant SSCs that alert personnel to the occurrence of a criticality accident and the need for immediate evacuation. The nuclear incident monitoring system alerts RBOF personnel to evacuate if a nuclear criticality (with fission product release) occurs in the facility. A remote alarm, located in the Building 221-H Control Room, is also activated to alert Building 221-H personnel to take action.

13.4.4.1 Nuclear Incident Monitoring System

Nuclear Incident Monitors (NIMs) are located where fuel handling activities are performed and where a criticality is considered a credible event. In addition to personnel protection in a criticality incident, the NIMs provide protection during fuel handling activities from a fuel raising event, or from a lowered basin water level condition. The RBOF is provided with six NIMs in three groups of two (see Chapter 2). The NIMs are designed to detect gamma radiation and to provide audible (alarm bells) and visual (flashing lights) alarms at a radiation rate between 0.5 and 3 R/hr. The alarm bells warn personnel to evacuate the vicinity in the event of a nuclear incident that could pose a personnel hazard. Alarm bells are located on each NIM chassis and at other places both inside and outside of the RBOF/RRF complex. Each NIM chassis has a red flashing light on top that activates with the audio alarm sound. The bell and light on the chassis indicate the direction of the incident so personnel can evacuate away from the scene of the criticality. The evacuation alarm response is a memorized action that is reviewed annually by all RBOF personnel.

Each NIM chassis has four status lights indicating the unit's operational condition. White, green, and yellow lights are located on the chassis front. The white light indicates that the NIM is operating on normal power. When the white light is out, the unit is operating on batteries. A green light indicates normal operation. When the NIM malfunctions, the green light goes out, an amber fault light is illuminated, and an intermittent audio indicator on the unit sounds. The red flashing light on top of the chassis indicates that the unit is in alarm condition. In addition to power and fault lights, NIMs have voltage and ampere meters to verify that the batteries and charging circuits are functioning properly. Procedures and training are provided for RBOF personnel monitoring the NIM system status.

A NIM status display is located in the RBOF Control Room on "Rack C." Indications of alarmed, inoperative, or bypassed NIMs are provided. Control room alarm response procedures and associated training are available to RBOF personnel. Table 13.4-1 lists the NIM trouble alarms displayed in the RBOF Control Room.

13.4.4.2 Remote 244-H General Alarm (Building 221-H)

A single annunciator plate and associated horn are provided in the Building 221-H Control Room to alert personnel of the following possible degraded condition in the RBOF and RRF:

- RBOF NIM trouble

When an alarm is received, Building 221-H personnel must be dispatched to the RBOF and RRF to determine which of the conditions caused the alarm. Procedures and training are provided to guide Building 221-H personnel response. Only the RBOF NIM trouble is safety significant. However, the safety-significant alarms cannot be distinguished from other alarms without going to the RBOF and RRF Control Rooms and inspecting the alarm panels.

The Remote 244-H General Alarm is only required by the TSR when the RBOF and RRF are unstaffed in a shutdown mode. When the RBOF and RRF are staffed, this alarm is bypassed to prevent it from alarming in the Building 221-H Control Room, since RBOF personnel will respond to alarms. The RBOF Control Room has an indicator light for the bypassed/unbypassed status of this alarm. The Building 221-H Control Room does not have bypassed/unbypassed indication. Formal, procedurally guided communication between the two control rooms is required to ensure the proper awareness of the change of bypassed/unbypassed state. Change of bypass status is accomplished with a key-activated switch in the RBOF Control Room. After the Building 221-H alarm is unbypassed, an alarm state in one of the monitored parameters is simulated. Receipt of the alarm in Building 221-H is then verified. Analysis of conditions in the Building 221-H Control Room is beyond the scope of this SAR.

13.4.4.3 Remote 244-H Nuclear Incident Alarm (Building 221-H)

A NIM alarm with an annunciator plate and horn is provided in the Building 221-H Control Room. The Building 221-H Control Room monitors the RBOF and performs the area emergency coordination function. This alarm is intended to be active at all times and cannot be bypassed, even when the RBOF is manned. Procedures and training for response to this alarm are provided. Analysis of conditions in the Building 221-H Control Room is beyond the scope of this SAR.

13.5 DESIGN OPTIMIZATION OF HUMAN-MACHINE INTERFACES

The RBOF/RRF is a Hazard Category 2 facility having both safety-class and safety-significant SSCs and programs. Several SSCs and programs are identified in the RBOF/RRF hazards and accident analysis that meet the safety criteria in DOE-STD-3009-94 (Ref. 10). The Functional Classification Report (Ref. 6) was reviewed and evaluated, as described in Section 13.3, to obtain a list of safety equipment and programs. Human actions in the RBOF affecting safety are those associated with monitoring facility parameters that are initial conditions of the SAR; controlling on irradiated fissile nuclear material type, amount, and location; and responding to criticality alarms for prompt evacuation. In the RRF, human actions affecting safety are primarily associated with following administrative controls on the type and location of hazardous materials and with ensuring proper status of spill control structures.

The Functional Classification Report and other sections were used to identify RBOF and RRF HMIs that control or monitor safety SSC status. Section 13.4 describes the safety HMIs. This description is the basis for the human factors evaluation in this section.

13.5.1 HUMAN FACTORS EVALUATION

The human factors evaluation was conducted using human factors checklists contained in WSRC-RP-95-460 (Ref. 7). Safety SSCs were evaluated as acceptable if the following human factors criteria were addressed:

- HMI design guidelines
- Physical work environment guidelines
- Organizational consideration guidelines

Each of the human factors criteria was determined to meet human factors guidelines if the system(s)/component(s) met the following criteria:

- Are designed to human factors related engineering standards that meet or exceed national standards
- Were subjected to a documented human factors study conducted by the system/component manufacturer
- Are a standard industrial design that has been used for an extended period of time in this or similar applications at SRS or at other facilities without significant or reoccurring problems
- Are the subject of facility or site-level commitments to comply with DOE Orders
- Were evaluated directly with respect to human factors guidelines using appropriate checklists or other techniques and were judged to meet the guidelines

If any of these five criteria is applicable, then the safety SSC is judged to meet the appropriate guidelines. If the safety SSC meets the human factors guidelines, then the SSC is evaluated to be capable of being safely and effectively operated.

HMI analysis used human factors checklists and DOE Order commitments as the assessment method. The physical work environment was evaluated by confirming the existence of design standards or SRS programmatic requirements. Organizational consideration analysis used SRS programmatic requirements and DOE Order commitments as the assessment method.

Table 13.5-1 lists the topic areas that were evaluated in each of these areas. The RBOF and RRF do not use computer systems as safety SSCs. The computer software interface topic area is listed for reference only. The human factors checklists and associated documentation are contained in S-CLC-E-00074 (Ref. 11).

13.5.2 EVALUATION CONCLUSIONS

HMIs for safety SSCs are evaluated as meeting human factors requirements. Company-level procedures controlling equipment identification, status boards, operator aids, and procedure validation are in place and applicable to the RBOF and RRF. The equipment identification procedure also references DOE human factors guidelines for label size and placement. These procedures and guidelines were assumed to have been properly applied for the purpose of this analysis.

Environmental aspects of the RBOF and RRF were designed to accepted industry standards, and programs exist to ensure that the design concepts are maintained. These standards have been used successfully at SRS for many years. During normal operations, the RBOF and RRF Control Rooms will not require protective clothing or personal breathing protection. The basin and cask handling areas require protective clothing (anti-contamination) during normal handling of fissile materials. Cask venting and decontamination operations, associated accident scenarios, as well as accident scenarios in the fuel handling area may require breathing protection, but operator recovery actions are not credited. Therefore, the impact of breathing protection for these activities and related human actions was not evaluated. F-FHA-H-00014 (Ref. 12) identified several minor deficiencies in emergency lighting. However, these deficiencies do not impact any safety-class or safety-significant actions discussed in this chapter.

Organizational consideration requirements are subject to DOE Order commitments and are assessed as fully adequate to support acceptable operator performance. Staffing levels and personnel work load are administratively controlled. The procedure guidelines require that a system be in place to ensure that appropriate procedures are written, the procedures are usable, and the procedures are verified and validated. The training guidelines require a system to be provided to ensure that the necessary training is developed, the training is properly presented, and the training recipients are tested to confirm mastery of the subject matter. For procedures and training requirements, site- and facility-level administrative controls are in place to comply with DOE Orders.

13.6 REFERENCES

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13.7 TABLES

Table 13.4-1 RBOF Control Room Alarms

Location	Alarm
<u>NIM Trouble Alarms</u>	
Storage and Cask Basin	NIM Trouble (one NIM Pair)
Inspection or Disassembly Basin	NIM Trouble (two NIM Pairs)

Table 13.5-1 Human Factors Evaluation Topic Areas (Ref. 11)

Human-Machine Interface	Physical Work Environment	Organizational Considerations
<u>System Hardware Interface</u> <ul style="list-style-type: none"> • Physical Access • Voice Communication Systems • Operational Aids • Controls • Instrumentation • Labels and Location Aids • Panel Layout • Control of Equipment and System Status • Allocation of Controls <u>Computer Software Interface</u> (Not used in RBOF/RRF analysis) <ul style="list-style-type: none"> • Character Size and Proportion • Coding Techniques • Enhancement Coding Dimensions • Labels • Abbreviations • Feedback • Display Density • Color • Frame Specifications • Frame Hierarchy and Elements • Menu Selection • Prompts • Dynamic Displays • Mimic Displays • Data Displays • Tabular Data • Graphic Displays • Alarms - Audible • Alarms - Data Messages 	<ul style="list-style-type: none"> • Heating, Ventilating and Air Conditioning • Work Space Lighting • Work Space Noise Levels • Work Space Physical Access • Personal Protective Equipment 	<ul style="list-style-type: none"> • Minimum Staffing Levels • Allocation of Control Functions • Overtime Restrictions • Communications Between Shifts • Procedures • Training

RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 14

QUALITY ASSURANCE

AUGUST 1997

UNCLASSIFIED
DOES NOT CONTAIN
UNCLASSIFIED CONTROLLED
NUCLEAR INFORMATION

ADC &
Reviewing
Official:

J.N. Dewes
J.N. DEWES, Manager, Regulatory Programs

Date: 9/23/97

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

SAVANNAH RIVER SITE

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SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

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ACRONYMS

DOE	Department of Energy
QA	Quality Assurance
RBOF	Receiving Basin for Offsite Fuels
RRF	Resin Regeneration Facility
SAR	Safety Analysis Report
SFSD	Spent Fuel Storage Division
SRS	Savannah River Site
WSRC	Westinghouse Savannah River Company

14.0 QUALITY ASSURANCE

14.1 INTRODUCTION

The purpose of this chapter is to provide information concerning Quality Assurance (QA) that satisfies Department of Energy (DOE) Order 5480.23, paragraph 8.b.(3)(r) (Ref. 1). Chapter 14 of the Generic Safety Analysis Report (SAR) (Ref. 2) addresses the Savannah River Site (SRS) QA Program, while this chapter addresses facility implementation of the site program, referencing the Generic SAR, as appropriate, for site-level information. This chapter includes descriptions of the following items:

- The QA Program and organization
- Document control and records management
- The QA process for ensuring that safety-related work satisfies performance criteria

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14.2 REQUIREMENTS

The Receiving Basin for Offsite Fuels (RBOF) and the Resin Regeneration Facility (RRF), as part of the Spent Fuel Storage Division (SFSD) at SRS, are required to comply with a number of DOE Orders, as well as codes, standards, and regulations, that govern QA policies and programs. Section 14.2 of the Generic SAR (Ref. 2) discusses these requirements.

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14.3 QUALITY ASSURANCE PROGRAM AND ORGANIZATION

The RBOF and RRF form a single facility that is part of SFSD. The SFSD QA Program complies with the requirements of Westinghouse Savannah River Company (WSRC) Procedure Manual 1Q (Ref. 3), as described in Section 14.3 of the Generic SAR (Ref. 2), which includes the following topics:

- Quality policy
- SRS QA Program definition
- WSRC organization and responsibilities
- Management processes
- Graded approach
- Special program requirements
- Personnel training and qualification
 - Qualification of personnel
 - Training
 - Training plans
 - Instructors

14.3.1 SFSD QA ORGANIZATION

WSRC Procedure Manual RD-1 (Ref. 4) describes the structure and interdisciplinary functions of the organizations within SFSD.

14.3.2 SFSD QA PROGRAM

The SFSD QA Program establishes the cognizant quality function for the organizations in the division. The QA Program elements for the division are as specified in WSRC Procedure Manual 1Q (Ref. 3). SFSD satisfies site-level requirements directly using Manual 1Q or lower-tier procedures that implement Manual 1Q requirements. A cross-reference of Manual 1Q requirements to division-level procedures is maintained in accordance with WSRC Procedure Manual R8.1 (Ref. 5). The SFSD QA organization performs its review functions in accordance with Manual R8.1 requirements.

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14.4 QUALITY IMPROVEMENT

Quality improvement at the RBOF and RRF is part of an ongoing effort at SRS. Section 14.4 of the Generic SAR (Ref. 2) details the quality improvement effort at SRS.

14.4.1 CONTROL OF NONCONFORMING ITEMS AND ACTIVITIES

Nonconforming items that affect programs or processes are reported on nonconformance reports. Nonconforming activities that affect programs or processes are reported on program deficiency reports. These reports are prepared in accordance with WSRC Procedure Manual 1Q (Ref. 3).

14.4.2 CORRECTIVE ACTIONS

The corrective action process is implemented through WSRC Procedure Manual 1Q (Ref. 3). Significant conditions adverse to quality are identified, reported, tracked, and corrected using this process.

14.4.3 LESSONS LEARNED AND TRENDING

The WSRC lessons learned policy is implemented through requirements in WSRC Procedure Manual 1B (Ref. 6). Quality-related information is monitored, evaluated, and reported for the purpose of trending activities through implementation of requirements in WSRC Procedure Manual 1Q (Ref. 3). WSRC Procedure Manual E7 (Ref. 7) specifies requirements for the identification, testing, collection, and analysis of performance data for systems and components to improve equipment reliability and availability through early detection of equipment degradation.

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14.5 DOCUMENTS AND RECORDS

As described in Section 14.5 of the Generic SAR (Ref. 2), WSRC Procedure Manual 1Q (Ref. 3) defines the responsibilities and requirements for the following activities relating to documents designated as QA records at SRS:

- Generation
- Identification
- Validation
- Receipt
- Indexing
- Storage
- Preservation
- Retrieval
- Correction
- Disposition

14.5.1 PROCEDURES AND WORK INSTRUCTIONS

SFSD administrative and technical procedures are prepared, reviewed, approved, and controlled in accordance with WSRC Procedure Manual RD-1 (Ref. 4). The technical procedure requirements in Manual RD-1 supplement the requirements in WSRC Procedure Manual 2S (Ref. 8) and do not relax any of the Manual 2S site-level requirements. The SFSD QA organization reviews procedures in accordance with WSRC Procedure Manual R8.1 (Ref. 5). Work instructions are controlled through WSRC Procedure Manual 1Y (Ref. 9).

14.5.2 DOCUMENT CONTROL

The Administration and Logistics Division provides document control services for SFSD. WSRC Procedure Manuals RD-1 and B14.1 (Ref. 4, 10) discuss SFSD activities and interfaces with the document control organizations. RBOF and RRF manuals and procedures are developed, reviewed, and approved in accordance with the SFSD requirements discussed in Section 14.5.1. Approved documents are transmitted to the document control organization for retention and distribution.

14.5.3 QA RECORDS

QA records are maintained to furnish documentation for the verification of activities and items affecting quality. WSRC Procedure Manual 1Q (Ref. 3) specifies site-level records retention requirements. SFSD records retention requirements are specified in WSRC Procedure Manual RD-1 (Ref. 4) and satisfy Manual 1Q requirements.

14.6 QUALITY ASSURANCE PERFORMANCE

14.6.1 WORK PROCESSES

Work processes are described in Section 14.6.1 of the Generic SAR (Ref. 2), which includes the following topics:

- Performance of work
- Identification and control of items
- Handling, storing, and shipping
- Calibration and maintenance of monitoring and data collection equipment
- Status indicators
- Control of computer software

Work is performed at SFSD facilities, which include the RBOF and RRF, in accordance with established controls defined in WSRC Procedure Manuals 1Q, 2S, and 1Y (Ref. 3, 8, 9).

14.6.2 DESIGN

Section 14.6.2 of the Generic SAR (Ref. 2) details the SRS Design Program and includes the following topics:

- Design change control
- Temporary modifications
- Design interfaces
- Design records
- Design verification

14.6.2.1 Design Control

SFSD Engineering is the cognizant technical function and design authority for the division, which includes the RBOF and RRF. The design authority uses WSRC Procedure Manuals E7 and E11 (Ref. 7, 11) as design authority guidance documents. WSRC Procedure Manual R5.1 (Ref. 12) provides additional implementing procedures for the design authority. The SFSD QA organization provides the required QA reviews of design activities in accordance with WSRC Procedure Manual R8.1 (Ref. 5).

14.6.2.2 Computer Software

The SFSD Quality Program for computer software provides assurance that, when software is used, the software satisfies the specified design and operating criteria. Software that affects items such as special nuclear material accountability and criticality safety is included in this program. Computer software requirements in WSRC Procedure Manual RD-1 (Ref. 4) incorporate the software inventory requirements in WSRC Procedure Manual 1B (Ref. 6) and the requirements for the evaluation of existing software in WSRC Procedure Manual 1Q (Ref. 3).

14.6.3 PROCUREMENT

Section 14.6.3 of the Generic SAR (Ref. 2) details the SRS Procurement Program and includes the following topics:

- Procurement Program definition
- Supplier selection and evaluation
- Product acceptance

SFSD implements requirements for the preparation, review, and approval of the following items through WSRC Procedure Manuals 1Q and 7B (Ref. 3, 13):

- Purchase requisitions
- Amendments to requisitions
- Purchase specifications
- Bid packages
- Other procurement documents

The SFSD QA organization reviews all Level 1 and 2 procurement documents in accordance with WSRC Procedure Manual R8.1 (Ref. 5) and performs related self-assessments in accordance with WSRC Procedure Manual 1B (Ref. 6). Requirements for the following QA Program elements are implemented through WSRC Procedure Manual R3.2 (Ref. 14):

- Control of purchased items and services
- Identification and control of items
- Packaging, handling, shipping, and storage

14.6.4 INSPECTION AND TESTING FOR ACCEPTANCE

Section 14.6.4 of the Generic SAR (Ref. 2) details the SRS Inspection and Testing for Acceptance Program and includes the following topics:

- Inspection
- Acceptance testing
- Measuring and test equipment

14.6.4.1 Inspections, Verifications, Self-Checks, Tests, and Operating Status

Inspection, verification, and self-checking requirements are met at appropriate points in the RBOF/RRF process in accordance with WSRC Procedure Manual 1Q (Ref. 3). Satisfying these requirements ensures conformance to applicable criteria.

SFSD has two categories for peer verification: in-process inspections and independent verifications. In-process inspections are performed concurrently with activity performance, while independent verifications are separated by time and/or distance. Personnel performing peer verifications are trained and qualified in accordance with Manual 1Q.

WSRC Procedure Manuals RD-1, 2S, and R3.2 (Ref. 4, 8, 14) also specify inspection, test, and operating status requirements.

14.6.4.2 Test Control

Test control requirements are met using WSRC Procedure Manual RD-1 (Ref. 4). SFSD Engineering reviews all test procedures and prescribes any test controls required over and above the Manual RD-1 requirements.

14.6.4.3 Control of Measuring and Test Equipment

Measuring and test equipment, as well as installed process instrumentation, is calibrated, controlled, and used in accordance with requirements in WSRC Procedure Manual 1Q (Ref. 3).

14.6.5 ASSESSMENT

Section 14.6.5 of the Generic SAR (Ref. 2) details the SRS Assessment Program and includes the following topics: management assessment and independent assessment.

SFSD implements WSRC Procedure Manual 1B (Ref. 6) requirements for performing assessments involving the measurement of item and service quality, the measurement of work performance, and the identification of improvement opportunities. The SFSD QA organization and RBOF/RRF management establish the scope and schedule for these self-assessments.

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 15

EMERGENCY PREPAREDNESS PROGRAM

AUGUST 1997

UNCLASSIFIED DOES NOT CONTAIN UNCLASSIFIED CONTROLLED NUCLEAR INFORMATION	
ADC & Reviewing Official:	<u>J.N. Dewes</u>
J.N. DEWES, Manager, Regulatory Programs	
Date:	<u>9/23/97</u>

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

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SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

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ACRONYMS

AEC	Area Emergency Coordinator
ALARA	As Low As Reasonably Achievable
DOE	Department of Energy
DOE-HQ	Department of Energy - Headquarters
EAL	Emergency Action Level
EDO	Emergency Duty Officer
EOC	Emergency Operations Center
EOF	Emergency Operations Facility
EPAP	Emergency Preparedness Administrative Procedure
EPIP	Emergency Plan Implementing Procedure
EPZ	Emergency Planning Zone
ERO	Emergency Response Organization
FEC	Facility Emergency Coordinator
FHA	Fire Hazards Analysis
JIC	Joint Information Center
LSR	Logistics Support Room
NIM	Nuclear Incident Monitor
OSC	Operations Support Center
PA	Public Address
RBOF	Receiving Basin for Offsite Fuels
RCO	Radiological Control Operations
RRF	Resin Regeneration Facility
SFSD	Spent Fuel Storage Division
SRS	Savannah River Site
SRSOC	Savannah River Site Operations Center
SST	Selective Signaling Terminal
TLD	Thermoluminescent Dosimeter
TSR	Technical Support Room
WSI-SRS	Wackenhut Services Inc. - Savannah River Site
WSRC	Westinghouse Savannah River Company

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15.0 EMERGENCY PREPAREDNESS PROGRAM

15.1 INTRODUCTION

The purpose of this chapter is to provide information on emergency preparedness that satisfies the requirements of Department of Energy (DOE) Order 5480.23, paragraph 8.b.(3)(s), as amplified in Attachment 1, paragraph 4.f.(3)(d)19, of the Order (Topic 19) (Ref. 1). This chapter also includes information that partially satisfies the requirements of DOE Order 5480.23, paragraphs 8.b(3)(b), (f), and (u), which are addressed in each chapter of the Safety Analysis Report (SAR) as necessary to provide explicit correlation to topics (Ref. 2).

This chapter describes the Savannah River Site (SRS) Emergency Preparedness Program as it relates to the Receiving Basin for Offsite Fuels (RBOF) and the Resin Regeneration Facility (RRF), which are part of the Spent Fuel Storage Division (SFSD), at SRS. Even though the RBOF and RRF are part of SFSD, the RBOF and RRF use the emergency planning that is in place for H-Area Separations.

The objective of this chapter is to demonstrate the commitment of the RBOF and RRF to the DOE Orders described in Section 15.2 of the Generic SAR (Ref. 3). The information provided by this chapter includes the following (Ref. 2):

- Identification of the scope of the facility emergency preparedness plan (i.e., the spectrum of emergencies encompassed)
- A description of the philosophy, objectives, organization, and emergency response of the facility Emergency Preparedness Program

Chapter 15 of the Generic SAR describes the overall site Emergency Preparedness Program. This chapter presents facility-specific information, referencing the Generic SAR, as well as other chapters, as appropriate.

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15.2 REQUIREMENTS

The RBOF and RRF follow Section 15.2 of the Generic SAR (Ref. 3), which provides the list of documents that specify codes, standards, regulations, DOE Orders, as well as other documents that govern the Emergency Preparedness Program.

In addition, the RBOF and RRF implement WSRC-SCD-7 (Ref. 4), which describes the emergency planning in place for H-Area Separations that is applicable to the RBOF and RRF.

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15.3 SCOPE OF EMERGENCY PREPAREDNESS

The emergency planning in place for H-Area Separations, which is applicable to the RBOF and RRF, provides for emergency response measures to ensure maximum protection for onsite and offsite populations, to limit damage to facilities and equipment, and to limit adverse impacts on the environment. This chapter addresses individual and organizational responses to all emergencies that could occur in the RBOF and RRF and that fall within the emergency classification system. The types of incidents that may cause emergencies in the RBOF and RRF include the following:

- Natural phenomena, such as tornadoes and earthquakes
- Accidental releases of radioactive or toxic chemicals
- Equipment failures or malfunctions
- Industrial accidents
- Security-related events, such as bomb threats and extortion

Of the various hazards that have been analyzed and identified for potential impact on H Area, the consequences of radiological hazards bound all other hazards (Ref. 5).

Section 15.3 of the Generic SAR (Ref. 3) summarizes accidents typically postulated for SRS facilities that can result in the release of significant amounts of radioactive and/or other hazardous materials and provides the complicating factors associated with the accidents.

The following sections summarize the postulated RBOF/RRF accidents, including hypothetical accidents with very low occurrence frequencies, that can result in the release of significant amounts of radioactive and/or other hazardous materials. Chapter 3 provides the consequences of, and further information on, the postulated RBOF/RRF accidents that have the potential for significant releases of radioactive and/or other hazardous materials.

15.3.1 NATURAL DISASTERS

Natural disasters with the potential to cause significant damage over the entire site (including the RBOF and RRF) include tornadoes, hurricanes, and earthquakes.

Section 15.3.1 of the Generic SAR (Ref. 3) provides discussions of tornadoes, hurricanes, and earthquakes and their complicating factors, which also apply to the RBOF and RRF. The accident analysis in Chapter 3 discusses accidents resulting from, or impacted by, natural disasters.

15.3.2 FIRES

There is a small probability that a fire could occur at the RRF from a spill of anion resins. This event assumes that the resin dries to a dry nitrate form (which is readily ignitable) and an ignition source is available. In addition, fire and explosion could also occur at the RRF in waste tank EP38 if the coolant of a received cask, when discharged to the waste tank, resulted in a flammable or explosive concentration of vapors in the tank. The rupture of the waste tank, caused by an explosion in the RRF, would lead to the release of contaminated liquid within the RRF.

Thorough quantitative Fire Hazards Analyses (FHAs) have been performed for the RBOF and RRF (Ref. 6), Support Facility Expansion (Ref. 7), and Rain Cover Addition (Ref. 8). These FHAs concluded that the RBOF/RRF complex is well maintained and meets the fire safety objectives of DOE Order 5480.7A (Ref. 9). The FHAs have concluded that a fire within the RBOF/RRF complex will pose no undue risk to the public or site employees because of the facility's noncombustible construction, the lack of continuity of combustibles in most areas, and the nature of the operation within the facility.

15.3.3 OPERATIONAL EMERGENCIES

The operational events considered in Chapter 3 include radionuclide release due to fuel cutting, liquid waste transfer error, spills, fuel rupture in storage, and nuclear criticality.

Operational events caused by equipment failures and/or human errors include the following:

- Fuel Cutting: The fuel core could be inadvertently cut in the repackaging basin, prior to bundling of the element, causing a release of airborne and/or high water activity to the work area.
- Liquid Waste: Liquid waste could be released either to high-level waste tank 21 or 23 or to the waste system for the Building 211-H Outside Facilities due to a transfer error.
- Spill at Hose Rack: Releases of contaminated water could occur at the hose rack platform during the handling of portable deionizers from the reactor areas.
- Fuel Rupture in Storage: Ascending bubbles in the basin water or changes in the appearance of cladding may be indicative of ruptured fuel. Fuel ruptures can contribute to basin water contamination and, in unusual circumstances, to a nuclear criticality accident.

The nuclear criticality accidents considered in Chapter 3 may be caused by the planned movement/manipulation of fuels in which infractions occur to the point that a critical mass of fuel is configured. These movements or manipulations can generally be grouped as follows:

- Storage Basin Operations: Storing fuel from the cask unloading basin, conducting traveling bridge operations, relocating fuel within the storage basin, and transferring fuel from the storage basin to a working basin can lead to a nuclear criticality accident.
- Cask Loading: Overloading the shipping cask, misloading and dropping the cask, and dropping fuel into the cask can lead to a nuclear criticality accident.
- Cask Unloading: Retrieving fuel from the cask, loading transfer buckets, and loading bundles can lead to a nuclear criticality accident.
- Working Basin Operations: Cropping fuel assemblies and placing buckets in isolation can lead to a nuclear criticality accident.

Chapter 3 discusses in more detail these activities and their corresponding consequences and probabilities.

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15.4 EMERGENCY PREPAREDNESS PLANNING

15.4.1 EMERGENCY RESPONSE ORGANIZATION

15.4.1.1 Normal Organization

The RBOF Operations Manager is assigned to the day shift and has overall responsibility for the day-to-day operations of the facility.

The shift supervisors are in charge of the operations in the facility. Operators are assigned to the shift supervisors to perform and monitor the operations.

The Building 221-H Shift Manager is normally the Area Emergency Coordinator (AEC) when an emergency is declared in H Area where the RBOF/RRF is not the incident facility. The Building 221-H Shift Manager may be relieved of AEC responsibility by a qualified senior management representative.

The RBOF/RRF Shift Supervisor is normally the Facility Emergency Coordinator (FEC) when an emergency is declared for the RBOF and RRF.

WSRC-SCD-7 (Ref. 4) describes additional area-level emergency personnel, including, but not limited to, the Operations Support Center (OSC) Coordinator, OSC Radiological Control Operations (RCO) Coordinator, and Protective Force Shift Lieutenant.

15.4.1.2 Onsite Emergency Response Organization

SITE (CENTRAL) EMERGENCY RESPONSE ORGANIZATION

The site-level Emergency Response Organization (ERO) includes personnel in the Emergency Operations Facility (EOF), Emergency Operations Center (EOC), Savannah River Site Operations Center (SRSOC), Technical Support Room (TSR), Logistics Support Room (LSR), and Joint Information Center (JIC). Section 15.4.1 of the Generic SAR (Ref. 3) describes in detail the site-level ERO, including functions and responsibilities of its personnel.

Figure 15.4-1 illustrates the overall organization responsible for responding to an emergency in the RBOF and RRF. Section 15.4.1 of the Generic SAR indicates the overall interfaces between various emergency response facilities, including the offsite facilities.

AREA EMERGENCY RESPONSE ORGANIZATION

The H-Area ERO for the RBOF and RRF consists of the AEC, FEC, RCO Coordinator, members of the emergency response teams, and the area security staff.

The Building 221-H Shift Manager is the AEC for H Area. The Building 221-H Control Room Supervisor replaces the Shift Manager as the AEC if the Shift Manager is unable to perform the AEC's duties.

The H-Area AEC has the authority to declare an emergency in H Area, including the RBOF and RRF. The main function of the AEC during an emergency in the RBOF or RRF is to provide overall direction and coordination of the area emergency response effort. WSRC-SCD-7 (Ref. 4) details other functions of the AEC during an emergency in the RBOF or RRF.

The primary responsibilities of the FEC is to form and direct emergency response teams to perform searches, first aid, corrective/mitigative actions, and other functions as directed by the AEC. The FEC is responsible for activating the OSC. WSRC-SCD-7 delineates other responsibilities of the FEC. A qualified member of the Building 221-H shift performs the duties of the AEC if the Building 221-H Control Room Supervisor is unable to perform the AEC's responsibilities. Building 707-H is the designated OSC for the RBOF and RRF.

The primary responsibilities of the RCO Coordinator are to report to the OSC and to assess hazardous material exposure hazards resulting from the RBOF/RRF emergency situation. The RCO Coordinator also forms and directs monitoring teams to perform surveys in the RBOF and RRF. WSRC-SCD-7 delineates other responsibilities of the RCO Coordinator. The RCO Shift Supervisor of Building 221-H is the designated RCO Coordinator for the RBOF and RRF.

The Protective Force Shift Lieutenant of H Area, as provided by Wackenhut Services, Inc. - Savannah River Site (WSI-SRS), assumes the role of Protective Force Shift Lieutenant during an emergency in the RBOF or RRF. The main function of the Protective Force Shift Lieutenant is to interface with the FEC and AEC to coordinate protective emergency response with the overall area emergency response. WSRC-SCD-7 delineates other responsibilities of this position.

FACILITY EMERGENCY RESPONSE ORGANIZATION

The RBOF and RRF report to the H-Area AEC during emergencies or evacuations. On-shift personnel in the RBOF and RRF provide the initial response to emergencies within the facility. This facility ERO is supplemented by Building 221-H ERO personnel and other facilities as needed.

The RBOF/RRF Shift Supervisor is the FEC for an emergency in the RBOF or RRF during hours of facility operation, while the Building 221-H Control Room Supervisor acts as the FEC during off-shift hours. A qualified member of the shift assumes FEC responsibility if the FEC is unable

to perform the FEC's duties. The Building 221-H OSC is used if the RBOF/RRF OSC (Building 707-H) is uninhabitable.

The first and foremost responsibility of the FEC is the safety and health of facility workers. In addition, the FEC has the responsibility of classifying the emergency event and providing prompt notification to the AEC and Emergency Duty Officer (EDO). The FEC directs the implementation of the appropriate facility protective action response (i.e., sheltering, relocation, evacuation, remain indoors) for the personnel in the RBOF and RRF and activates the OSC. WSRC-SCD-7 (Ref. 4) delineates other responsibilities of the FEC.

The RCO Supervisor on shift in the RBOF and RRF assumes the duties and responsibilities of the facility RCO Coordinator. The facility RCO Coordinator may be relieved of duties by a senior RCO management representative.

A qualified member of the RBOF/RRF staff serves as the Incident Scene Coordinator during an emergency in the facility.

OTHER EMERGENCY RESPONSE ORGANIZATION POSITIONS

Other emergency response positions may be assigned to either area (H Area) or facility (RBOF/RRF) personnel, depending upon availability, by the responsible authority during an emergency in the RBOF or RRF. WSRC-SCD-7 (Ref. 4) delineates the responsibilities of these positions. These positions are as follows:

- Communicator/Logkeeper: Communicates the emergency information accurately and records critical information by time and date
- Accountability Coordinator: Identifies any missing personnel and advises the FEC/AEC
- Rally Point Coordinator: Coordinates protective actions for accountability of personnel at the rally point with the AEC, FEC, EDO, and the Protective Force Shift Lieutenant

15.4.1.3 Offsite Emergency Response Interactions

Section 15.4.1 of the Generic SAR (Ref. 3) discusses the responsibilities for interactions with offsite EROs.

15.4.2 ASSESSMENT ACTIONS

This section identifies the emergency classification system used at the RBOF and RRF and relates the classification system to the emergencies arising from the accidents, natural disasters, or security incidents. The notifications, coordination, and information to be communicated are

also summarized. The source terms and release rates, as well as identifiers of the consequence assessment module and plume methodologies, are specified in S-EHA-H-00003 (Ref. 5).

15.4.2.1 Classification System

During hours of facility operation, the RBOF/RRF Shift Supervisor is notified of the event first so that immediate action can be taken. The RBOF/RRF Shift Supervisor reviews the situation as the FEC using the Emergency Action Level (EAL) guidelines specified in WSRC-SCD-7 (Ref. 4). During off-shift hours, the Building 221-H Control Room Supervisor acts as the FEC. If the event parameters meet or exceed the EAL limits, the FEC determines the appropriate emergency classification. Upon notification of the event and emergency classification from the FEC, the Building 221-H Shift Manager assumes the duties and responsibilities of the AEC. The FEC also notifies the EDO of the event and emergency classification.

For classifying emergencies, the RBOF and RRF use the Emergency Plan Implementing Procedures (EPIPs) specified in WSRC-SCD-7 for H-Area Separations. Section 15.4.2.1 of the Generic SAR (Ref. 3) describes the classification system used for areas where classification procedures are not in place and for the general site.

15.4.2.2 Hazardous Material Emergencies

Section 15.4.2 of the Generic SAR (Ref. 3) provides the emergency classification categories and their descriptions. The RBOF and RRF also use these same categories, which are as follows:

- Notice of Unusual Event
- Alert
- Site Area Emergency
- General Emergency

15.4.2.3 Classification Responsibilities

When the event parameters in the RBOF and RRF reach and exceed the EAL limits, as provided by the EPIPs specified in WSRC-SCD-7 (Ref. 4) for H-Area Separations, the FEC (see Section 15.4.2.1) determines the appropriate emergency classification. Upon receiving notification of the event and the emergency classification from the FEC, the Building 221-H Shift Manager assumes the duties and responsibilities of the AEC. The FEC also notifies the EDO of the event and emergency classification.

The FEC/AEC activates the Emergency Control Room in accordance with the EPIPs specified in WSRC-SCD-7. The OSC Coordinator activates the OSC. The site-level ERO activates the EOC and TSR.

Once the TSR is activated, the emergency classification duty transfers to the TSR Manager. The FEC/AEC continues to evaluate and recommend changes in classification to the TSR Manager, as appropriate.

The EDO is responsible for classifying emergencies for the general site until the EOC is activated. Once the EOC is activated, the EOC Manager assumes this responsibility.

15.4.3 NOTIFICATION

SRS, including the RBOF and RRF, implements a notification process that has been pre-arranged and/or standardized to reduce any confusion during an emergency. Section 15.4.3 of the Generic SAR (Ref. 3) provides a listing of items that are included in the notification process.

RBOF and RRF personnel are responsible for notifying the RBOF/RRF Shift Supervisor upon observation of an abnormality. The FEC is responsible for notifying the EDO and AEC of the event, status, and classification of the emergency as delineated in WSRC-SCD-7 (Ref. 4).

RBOF and RRF personnel are notified through alarms and/or Public Address (PA) announcements. Fission product detection systems (gamma radiation monitors or Nuclear Incident Monitors [NIMs]) are also installed in the facility. These monitors provide visual indications locally if pre-set radiation levels are exceeded and also provide prompt evacuation alarms that alert facility personnel in the event of a nuclear criticality incident coupled with a release of fission products.

15.4.3.1 Offsite Notifications

The SRSOC is responsible for ensuring that offsite authorities are notified and for initiating a notification call to the on-call Site Environmental Protection Coordinator to meet environmental regulatory requirements. Section 15.4.3 of the Generic SAR (Ref. 3) provides additional information on the responsibilities of the Site Environmental Protection Coordinator.

15.4.3.2 Notification of State/Local Authorities for Events That Involve Radiological Material Releases

Section 15.4.3 of the Generic SAR (Ref. 3) describes notification of state/local authorities and the method of communication for events involving radiological material releases.

15.4.3.3 Notification of State/Local Authorities for Events That Do Not Involve Radiological Material Releases

For events that do not involve or impact nuclear processes but may affect the offsite population, procedures have been established to use commercial telephones to provide prompt notification of

the emergency directly to the appropriate officials in the affected counties and states. Section 15.4.3 of the Generic SAR (Ref. 3) identifies information to be provided in the notification call for chemical-related events. This information is in accordance with the requirement of the Emergency Planning and Community Right-to-Know Act.

15.4.3.4 Other Agencies

The EDO or EOC Manager, as appropriate, notifies the following organizations of an emergency by commercial telephone: the Railroad Company, the Federal Aviation Administration, and the U.S. Coast Guard stationed at Savannah, Georgia. Section 15.4.3 of the Generic SAR (Ref. 3) identifies additional federal agencies that are notified during an emergency.

15.4.3.5 General Population

Residents are notified of an emergency by radio, television, sirens, and vehicles with PA capabilities. Section 15.4.3 of the Generic SAR (Ref. 3) provides details on the process for notifying the general population.

15.4.3.6 Onsite Notifications

The FEC is responsible for notifying the following personnel of incidents occurring within the RBOF and RRF:

- AEC and EDO (in the SRSOC)
- Personnel in the RBOF and RRF

The AEC is responsible for notifying personnel in H Area.

The FEC/AEC is responsible for notifying personnel in the RBOF and RRF by way of the safety alarm system and PA system. Facility/area ERO personnel are notified by the PA system or by activation of the pagers worn by key personnel. In addition to the safety alarm system, fire alarms and radiation monitors are provided in the RBOF and RRF to warn personnel in the facility of fires and nuclear criticality events.

The EDO is notified over the Selective Signaling Terminal (SST) system. If the SST is not available, notification may be made by way of the SRS telephone system or the emergency radio system.

Communications between H-Area personnel and other FECs in the H-Area Control Room and the S-Area Control Room may be made through the use of ring-down telephones. The SRS telephone system and emergency radio system serve as backup means of communication. Section 15.4.3 of the Generic SAR (Ref. 3) provides additional detail on onsite communication.

15.4.3.7 Site Emergency Response Organization

Section 15.4.3 of the Generic SAR (Ref. 3) describes the requirements for the limited and full activation of the site ERO. The chapter also describes the responsibilities of the SRSOC regarding the ERO.

15.4.3.8 DOE Assets

Requests for DOE assets are made through the Department of Energy - Headquarters (DOE-HQ) EOC.

15.4.3.9 Notification of DOE Field Personnel and DOE-HQ EOC

Section 15.4.3 of the Generic SAR (Ref. 3) describes the process for notifying DOE field personnel and the DOE-HQ EOC when the SRSOC receives an emergency.

15.4.4 EMERGENCY FACILITIES AND EQUIPMENT

15.4.4.1 Emergency Facilities

Site-level emergency facilities have been pre-designated and equipped with emergency equipment for use in an emergency. Site-level emergency response facilities include, but are not limited to, the EOC, SRSOC, TSR, LSR, and JIC. Section 15.4.4 of the Generic SAR (Ref. 3) describes the site-level emergency facilities and the equipment required to respond to an emergency in any SRS facility.

Emergency facilities applicable to H Area and available to the RBOF and RRF are as follows:

- Building 707-H OSC (Primary)
- H-Area Separations OSC (Alternate)
- Building 221-H Control Room
- Building 704-H Medical Station
- Building 221-H Emergency Decontamination Stations
- Building 221-H Decontamination Room
- Building 224-H Truck Port Vehicle Decontamination Point
- H-Area Security Facility

The H-Area Separations OSC, located on the first level of Building 221-H, is used if the RBOF/RRF OSC (Building 707-H) is uninhabitable. Equipment for the control room includes

normal SRS telephones and ring-down telephones. Additionally, the control room is equipped with the following communication equipment to communicate inside the RBOF, RRF, and H Area and with the EDO and TSR staff:

- SST telephone
- Telephones
- Hand-held radios

The OSC serves as an assembly point for emergency response personnel. The OSC is used to dispatch and coordinate the emergency response team activities (e.g., search, first aid, equipment maintenance, and equipment operations) in the RBOF and RRF. SRS telephones and radios are available for emergency response teams to communicate with each other and with the OSC. Equipment and supplies needed by the emergency response teams during an emergency are located near the OSC.

The medical station, located in Building 704-H, is staffed by Medical Department personnel and is available to provide medical services to RBOF and RRF personnel, in case of an emergency during day shifts (Monday through Friday). If an emergency in the RBOF or RRF occurs outside of these hours, then the F-Area medical staff is available to provide such services to RBOF and RRF personnel. Adequate equipment and supplies are available and maintained in these medical stations as determined by the Medical Department. In addition, emergency kits are available and maintained in the RBOF and RRF. The cabinets in Building 704-H are inventoried and verified in accordance with Emergency Preparedness Administrative Procedures (EPAPs) specified in WSRC-SCD-7 (Ref. 4). The H-Area first aid teams are available on all shifts (minimum of five members per shift) to provide emergency first aid in the RBOF and RRF.

There are two emergency decontamination stations in Building 221-H; both are located on the second level, one at the north end and the other at the south end. The personnel decontamination room is located in the RCO Count Room. A vehicle decontamination station is located in the Building 224-H Truck Port.

Building 221-H has a protective force facility to accommodate the area security personnel. WSI-SRS maintains security facilities, including access control points and central alarm stations. Communication equipment is available at all of these locations for coordination of the protective force personnel and for communications with other facilities including the SRSOC and Building 221-H Control Room.

15.4.4.2 Emergency Equipment

WSRC-SCD-7 (Ref. 4) describes the emergency equipment available to H Area and the RBOF and RRF during an emergency.

COMMUNICATIONS EQUIPMENT

- PA System: The area PA system provides a means of RBOF, RRF, and H-Area communications. The system can be activated from selected control rooms, with backup capability for activation provided from the protective force headquarters in H Area. The Building 221-H Control Room can override other users of the PA paging system.
- Telephone Systems:
 - The SST system is a dedicated ring-down telephone system linking over 90 SRS locations. It is the primary system for communications between the Building 221-H Control Room and the SRSOC. It is used as the primary means for sitewide notification of emergency conditions.
 - The SRS telephone system provides service to the entire site. It serves as the primary means of communication between the Building 221-H Control Room and the OSC. It is also a backup means of communication between the area emergency facilities and the site-level emergency response facilities and personnel.
 - The Bell Intercom System is a subset of the SRS telephone system. The system is available in an emergency to facilitate communications between the OSC and the Building 221-H Control Room.
 - A Bell Labs 334 System, installed in H Area, allows the RBOF to communicate with the AEC. The AEC has the capability of initiating communication with the RBOF.
 - A dedicated automatic ring-down telephone is in place between the RBOF shift supervisors office, the basin area and Building 221-H Control Room.
- Radio Systems: Hand-held radios are available for communications between RBOF, RRF, and H-Area emergency response teams; the Building 221-H Control Room; and the OSC in the event that the telephone systems are unavailable.

HEAVY CONSTRUCTION EQUIPMENT

Heavy construction equipment, if needed to mitigate the consequences of a natural phenomenon (e.g., earthquake, straight winds, or tornado) or some other emergency (e.g., explosion), may be requested through the EDO or TSR staff.

DECONTAMINATION EQUIPMENT

Personnel decontamination stations and equipment are available in two locations in Building 221-H (second level, north and south ends) and one location in the Building 221-H

RCO Count Room. These stations and equipment are maintained by RCO in accordance with Procedure 136 of WSRC Procedure Manual 5Q1.2 (Ref. 10).

FIRST AID TEAM EQUIPMENT

Equipment provided for use by the emergency response teams, first aid teams, or other emergency workers is contained in cabinets in the RBOF and RRF as well as in H Area. The cabinets contain such items as the following:

- Leather, rubber, and acid gloves
- Flashlights and batteries
- Hard hats
- Bolt cutters
- Tape
- Rope
- PC coveralls
- Safety belts
- AIDS kits

A complete list of the contents may be found in procedure 138-9-H-202 (Ref. 11). The EPAPs specified in WSRC-SCD-7 (Ref. 4) cover equipment cabinet locations and inventory frequencies.

SANITATION AND SURVIVAL EQUIPMENT

Sanitation and survival equipment, if needed, is requested through the EDO or TSR staff.

TRANSPORTATION EQUIPMENT

Other than emergency vehicles equipped for first aid team use, normal-use government vehicles, and vehicles assigned to H-Area facilities, all emergency transportation is requested through the EDO or TSR staff.

PERSONAL PROTECTIVE EQUIPMENT

The two self-contained breathing apparatuses available for use by RBOF and RRF are found outside of the 707-H building.

HAZARDOUS MATERIAL DETECTION AND MONITORING

WSRC-SCD-7 (Ref. 4) describes the portable, hand-held instruments available to monitor for hazardous material releases in the RBOF, RRF, and H Area.

DAMAGE CONTAINMENT EQUIPMENT

Damage containment equipment is requested through the EDO or TSR staff.

FIREFIGHTING EQUIPMENT

Fire extinguishers are located throughout the RBOF and RRF as described in the FHAs for the RBOF and RRF (Ref. 6, 7, 8).

Fire extinguishers located at the RBOF and RRF are inspected monthly by RBOF/RRF operations personnel in accordance with facility ticklers (i.e., a surveillance test database/scheduling mechanism used to generate and distribute surveillance notices and to track and record surveillance activities). Annual inspections are performed by the Fire Protection Division.

STANDBY/BACKUP POWER EQUIPMENT

WSRC-SCD-7 (Ref. 4) describes the need to determine the size and configuration of standby/backup power supplies for the RBOF and RRF. The RBOF and RRF facilities receive normal site power from Waste Management or standby power from the 254-3H diesel generator. The Waste Management diesel generator automatically starts when H Area loses power. When the diesel starts, power to Waste Management and the RBOF is restored. RBOF standby power is limited to those selected loads connected to distribution panel EP #513.69 and two air sample pumps. If necessary, portable power units can be requested through the EDO or TSR staff.

LOGISTIC SUPPORT EQUIPMENT

Controlled copies of procedures are maintained in storage cabinets of the Building 221-H Control Room and OSC.

15.4.5 PROTECTIVE ACTIONS

This section deals with equipment, procedures, and response actions at the RBOF and RRF necessary to provide maximum protection for facility personnel.

For hazardous material and radiological emergencies, protective actions are designed to keep onsite and offsite exposures as low as possible. This goal is accomplished by minimizing time

spent in the vicinity of the hazard, keeping as far from the hazard as possible, and taking advantage of available shelter.

For emergencies based on safeguards and security events, protective actions are designed to provide maximum physical safety of facility personnel while maintaining the operational and security integrity of the RBOF and RRF.

15.4.5.1 Onsite Precautionary Actions

Section 15.4.5 of the Generic SAR (Ref. 3) provides precautionary actions to be taken by the FEC/AEC and EDO. At the RBOF and RRF, these precautionary actions are handled by the Shift Manager/Supervisor if the RBOF/RRF is the incident facility.

15.4.5.2 Onsite Protective Response

The FEC/AEC notifies RBOF, RRF, and H-Area personnel to take appropriate action by using the safety alarm system and PA announcements. The SRS telephone system and personnel contact serve as backup means of notification of facility personnel.

The following sections discuss the protective actions that may be taken during an emergency.

REMAIN INDOORS

This protective action is specific for toxic gas releases posing an immediate danger. Nonessential personnel within a facility will remain inside. Nonessential personnel outdoors will enter the nearest building regardless of the type of structure. Outside doors and windows will be closed, and ventilation systems that draw in outside air will be shut down, unless such action creates a health and safety hazard. Essential personnel will report to their assigned emergency response locations, unless they are directed otherwise or unless it is unsafe to do so.

SHELTERING

Sheltering is the movement of facility/area personnel to the nearest pre-designated shelter locations with all doors, windows, and outside air intakes closed. This protective action is normally appropriate when radiological releases are of short duration or when relocation requires travel through the plume. Sheltering is also appropriate for some types of natural phenomena, such as when high wind or tornado warnings are issued.

The RBOF/RRF is an approved shelter location. WSRC-SCD-7 (Ref. 4) lists other approved shelter locations in H Area and available to H-Area personnel.

EVACUATION

Evacuation requires movement of facility/area personnel to a rally point or assembly area for accountability and, if warranted, relocation purposes.

The pre-designated evacuation rally points for the RBOF and RRF, are as follows:

- Primary rally point: H-Area main parking lot
- Alternate rally point: H-Area north parking lot
- RBOF/RRF-specific rally point: West of Building 706-H near fire alarm pull box

RELOCATION

Nonessential personnel will be directed by PA announcement to proceed to one of the evacuation rally points, and from there personnel will be transported out of the area to another SRS location. Relocation is normally directed by the EDO or EOC Manager in coordination with the AEC/FEC. H-Area essential personnel will report to their assigned emergency response locations unless otherwise directed.

POTASSIUM IODIDE

Section 15.4.5 of the Generic SAR (Ref. 3) discusses the specific situations when potassium iodide will be used and the responsibilities for administering the potassium iodide.

PERSONNEL ACTIONS FOR SECURITY-RELATED INCIDENTS

Protective actions for a security event will be determined by the Protective Force Shift Lieutenant in H Area and will be transmitted to the AEC/FEC for implementation.

EXPOSURE CONTROL

In an emergency, nonessential workers will remain indoors, take shelter, or relocate to another area in support of the As Low As Reasonably Achievable (ALARA) concept.

Radiological Exposures - Essential Workers

Section 15.4.5 of the Generic SAR (Ref. 3) specifies the badging, dosimetry, and Thermoluminescent Dosimeter (TLD) requirements for essential workers and the responsibilities for approving emergency response-related activities that involve radiological overexposures or substantial risk. The Generic SAR section also specifies requirements for providing a 24-hour capability to read TLDs in an emergency.

Chemical Exposures - Essential Workers

Section 15.4.5 of the Generic SAR (Ref. 3) describes the criteria for determining the emergency classification for events involving an actual or potential release of toxic chemicals and the responsibilities for initiating protective actions, including evacuation and use of protective clothing and respirators.

Emergency Response Facility Habitability

Section 15.4.5 of the Generic SAR (Ref. 3) discusses the responsibilities for deciding when to evacuate from an emergency response facility.

15.4.5.3 Decontamination

Section 15.4.5 of the Generic SAR (Ref. 3) specifies the requirements for monitoring all potentially contaminated personnel in the event of a radiological or toxic release. The Generic SAR section further specifies the requirements for decontamination and treatment of personnel in the emergency decontamination stations, decontamination room, and medical decontamination and treatment unit and discusses the use of the vehicle decontamination points.

Section 15.4.4 discusses the decontamination facilities in the RBOF, RRF, and H Area.

15.4.5.4 Access Control

Section 15.4.5 of the Generic SAR (Ref. 3) specifies the access control provisions onsite and at the site perimeter during potentially hazardous material releases or security-related events.

15.4.5.5 Responsibility for Protective Actions

The FEC is responsible for determining the initial protective actions in the RBOF and RRF. The FEC is also responsible for coordinating protective actions for the facility with the AEC, who is responsible for protective actions in H Area. These facility/area protective actions are performed in accordance with the EIPs specified in WSRC-SCD-7 (Ref. 4). Onsite protective actions for personnel in the RBOF, RRF, and H Area are intended to be used any time the FEC/AEC determines such actions are needed to protect these personnel.

Section 15.4.5 of the Generic SAR (Ref. 3) specifies responsibilities for protective actions beyond H Area.

15.4.5.6 Protective Equipment and Supplies

Section 15.4.4 describes protective equipment and supplies for the RBOF and RRF. Section 15.4.5 of the Generic SAR (Ref. 3) specifies other requirements for protective equipment and supplies for essential workers.

15.4.5.7 Protective Action Guides

Section 15.4.5 of the Generic SAR (Ref. 3) discusses onsite and offsite protective action guides, which also apply to the RBOF and RRF.

15.4.5.8 Emergency Response Planning Guidelines

Section 15.4.5 of the Generic SAR (Ref. 3) discusses Emergency Response Planning Guidelines developed by the American Industrial Hygiene Association for use as indicators of the need for protective actions based on chemical concentrations. These guidelines have been adopted by DOE for use in DOE contract sites, including SRS and the RBOF and RRF.

15.4.5.9 Records

The AEC/FEC and the communicators/logkeepers record and maintain accurate, complete, and chronological entries in their shift logs by following the appropriate EPIPs specified in WSRC-SCD-7 (Ref. 4). The shift log is maintained as an official record in accordance with the Sitewide Records Inventory and Disposition Schedule (Ref. 12).

Section 15.4.5 of the Generic SAR (Ref. 3) specifies requirements for other key positions to maintain other logs/documents and describes the responsibility of the WSRC Emergency Management Department of the Security and Emergency Preparedness Division to collect these logs/documents at the end of the event and to process them for records storage. The Generic SAR section also specifies recordkeeping requirements for activities related to emergency chemical and radiological exposures and medical treatments.

15.4.5.10 Personnel Accountability

Personnel at the RBOF and RRF are notified of an emergency by way of the PA system. The FEC is required to report the personnel accountability results to the H-Area AEC, who coordinates support responses as necessary to search for missing personnel.

Accountability for the RBOF and RRF is also accomplished by designated assembly points for each work discipline at the rally point so that supervision can account for their personnel. The results are reported to the Rally Point Coordinator, who then informs the AEC of the accountability status.

Additionally, personnel are accounted for when an operating facility is evacuated or when it becomes necessary to relocate all personnel in the area to another area.

Personnel accountability in the RBOF and RRF is conducted in accordance with the applicable requirements of WSRC-SCD-7 (Ref. 4).

15.4.5.11 Emergency Planning Zones

Emergency Planning Zones (EPZs) are areas for which planning is needed to ensure that prompt and effective actions can be taken to protect onsite personnel, public health and safety, and the environment.

In coordination with state officials, two EPZs have been established. Section 15.4.5 of the Generic SAR (Ref. 3) describes plume exposure and ingestion exposure EPZs and discusses the responsibility and determination of protective action recommendations, coordination, sheltering, and evacuation.

15.4.5.12 Communication

The RBOF and RRF use the PA system, telephone system, and radio system as means of communication during emergencies. WSRC-SCD-7 (Ref. 4) describes the communication equipment available at H Area (including the RBOF and RRF) during emergencies.

15.4.5.13 Termination

Each time the status of the emergency changes, onsite and offsite protective action recommendations are reviewed and modified accordingly. The same systems used to announce initial protective actions are used if changes in protective actions need to be communicated.

If site-level facilities have not been activated, the FEC and AEC use the EPIPs specified in WSRC-SCD-7 (Ref. 4) to terminate from an emergency. If site-level facilities have been activated, then it is the responsibility of the Emergency Director of the EOF to downgrade or terminate from the emergency.

15.4.5.14 Shutdown of Operations

The RBOF/RRF Shift Supervisor is responsible for the safe operational shutdown of the RBOF and RRF using Normal Operating Procedure 244-H-8000 (Ref. 13). Section 15.4.5 of the Generic SAR (Ref. 3) provides information on the authority for implementing safe shutdown procedures and the responsibility for ensuring the accomplishment of safe shutdown.

15.4.6 TRAINING AND EXERCISES

15.4.6.1 Training

The RBOF and RRF implement the training requirements of WSRC-SCD-7 (Ref. 4). The EPAPs specified in WSRC-SCD-7 provide initial and annual requalification training requirements for ERO members and include a listing of courses required by all ERO members.

Section 15.4.6 of the Generic SAR (Ref. 3) defines the Emergency Management Training Program and administration requirements for all personnel at SRS (including the RBOF and RRF).

15.4.6.2 Drills and Exercises

Drills and exercises are conducted to develop, maintain, and test response capabilities of ERO members and to validate the adequacy of emergency facilities, equipment, procedures, and training.

The RBOF and RRF implement the requirements of WSRC-SCD-7 (Ref. 4) for drills and exercises. The manual provides the types and frequencies of drills and exercises and the organization responsible for conducting drills and exercises.

15.4.7 REENTRY AND RECOVERY

15.4.7.1 Emergency Downgrade and Termination

Section 15.4.7 of the Generic SAR (Ref. 3) specifies the responsibilities and the criteria for downgrading or terminating an emergency, which are also applicable to the RBOF and RRF. The RBOF/RRF Shift Supervisor can initiate the termination of a Notice of Unusual Event-level emergency if that individual is the classifying authority.

15.4.7.2 Reentry

Section 15.4.7 of the Generic SAR (Ref. 3) provides information applicable to the RBOF and RRF on how emergency reentries are performed and what governs these emergency reentries.

Additionally, the RBOF and RRF use the EPIPs specified in WSRC-SCD-7 (Ref. 4) for performing emergency reentry into the facility.

15.4.7.3 Recovery

Recovery includes those actions necessary to restore an incident site and surrounding environs to a pre-emergency or safe condition.

The H-Area Separations Program Manager or designee is responsible for coordinating the recovery efforts with the SRS recovery organization in accordance with the requirements in the Site Overview provided in WSRC-SCD-7 (Ref. 4). All departments and support groups, including those at the RBOF and RRF, would assist the Separations Program Manager as necessary during the recovery.

15.5 REFERENCES

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2. Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports. DOE-STD-3009-94, U.S. Department of Energy, Washington, DC, July 1994.
3. Generic Safety Analysis Report - Emergency Preparedness Program, Chapter 15. G-SAR-G-0015, Rev. 0, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC, March 1995.
4. SRS Emergency Plan (U). WSRC-SCD-7, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.
5. Emergency Preparedness Hazards Assessment for the RBOF/RRF Facility (U). S-EHA-H-00003, Rev. 0, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC, September 1995.
6. Fire Hazards Analysis for RBOF (U). F-FHA-H-00014, Rev. 0, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC, June 1994.
7. Fire Hazards Analysis for Project S-5891, 244-H, Support Facility Expansion (U). F-FHA-H-00027, Rev. 0, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC, November 1995.
8. Fire Hazards Analysis for Project S-017, 245-H, Rain Cover Addition (U). F-FHA-H-00025, Rev. 0, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC, August 1995.
9. Fire Protection. DOE Order 5480.7A, U.S. Department of Energy, Washington, DC, February 1993.
10. Health Protection Department Radiation Survey Procedures (U). WSRC Procedure Manual 5Q1.2, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.

Procedure 136, "Site RCO Emergency Equipment Audit," Rev. 4, February 1997.
11. Emergency Equipment Checklist (U). Procedure 138-9-H-202, Rev. 2, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC, February 1997

12. Sitewide Records Inventory and Disposition Schedule (RIDS) (U). WSRC-IM-93-0060, Rev. 6, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC, October 1995.
13. RBOF Security Emergency Response (U). NOP 244-H-8000, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.

15.6 FIGURES

Figure 15.4-1 Normal and Emergency Response Organization Relationships for the RBOF, RRF, and H Area

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 16

PROVISIONS FOR DECONTAMINATION AND DECOMMISSIONING

AUGUST 1997

UNCLASSIFIED
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ADC &
Reviewing
Official: J.N. Dewes

J.N. DEWES, Manager, Regulatory Programs

Date: 9/23/97

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

SAVANNAH RIVER SITE

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SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

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ACRONYMS

CFR	Code of Federal Regulations
D&D	Decontamination and Decommissioning
DOE	Department of Energy
DOE-DP	Department of Energy - Defense Programs
DOE-EM	Department of Energy - Environmental Management
DOT	Department of Transportation
HVAC	Heating, Ventilation, and Air Conditioning
MC&A	Material Control and Accountability
NRC	Nuclear Regulatory Commission
RBOF	Receiving Basin for Offsite Fuels
RC&HP	Radiological Control and Health Physics
RRF	Resin Regeneration Facility
SAR	Safety Analysis Report
SFSD	Spent Fuel Storage Division
SNM	Special Nuclear Material
S/RID	Standards/Requirements Identification Document
SRS	Savannah River Site
SWM	Solid Waste Management
TSD	Treatment, Storage, or Disposal
WSRC	Westinghouse Savannah River Company

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16.0 PROVISIONS FOR DECONTAMINATION AND DECOMMISSIONING

16.1 INTRODUCTION

16.1.1 OBJECTIVE

The purpose of this chapter is to provide information that satisfies the requirements of Department of Energy (DOE) Order 5480.23, paragraph 8.b.(3)(t), as amplified in Attachment 1, paragraph 4.f.(3)(d)20, of the Order (Topic 20) (Ref. 1). This chapter also includes information, as applicable, that partially satisfies the requirements of DOE Order 5480.23, paragraphs 8.b.(3)(b), (f), and (u), as discussed in detail in DOE-STD-3009-94 (Ref. 2).

16.1.2 SCOPE

This chapter provides information on the provisions for Decontamination and Decommissioning (D&D) of the Receiving Basin for Offsite Fuels (RBOF) and the Resin Regeneration Facility (RRF). The scope of this chapter includes the following:

- Identification of the requirements for D&D provisions
- Description of the design features incorporated into major modifications of the RBOF and RRF to facilitate future D&D of the facility
- Description of the operational considerations to facilitate future D&D of the facility
- Description of the conceptual D&D plan

The graded approach to the level of detail provided in this chapter was influenced by the following:

- The Hazard Category 2 classification of the facility
- The moderate complexity of the facility
- The fact that the facility was completed and had operated for many years prior to the requirement to prepare the Safety Analysis Report (SAR)
- The facility's continuing mission of operation rather than D&D

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16.2 REQUIREMENTS

Requirements for D&D provisions are promulgated by federal and state regulators and industry professional associations. Requirements exist that cover several aspects of D&D, including provisions for D&D in facility design, operation, maintenance, and inspection; planning for D&D; and actual performance of D&D activities.

The majority of criteria regarding facility design and operation occur in general design documents rather than in ones specifically addressing D&D. This statement is logical when considering the fact that many good design practices also contribute to low contamination levels, a decrease in hot spots within systems, and ease of maintenance. For example, the design criteria for electrical conduit systems help prevent the conductors from being exposed to moisture and also help prevent radioactive liquids from being trapped in the conduit.

Codes and standards used by the various engineering disciplines in the course of specifying design, operation, maintenance, and inspection criteria are specified in other chapters, primarily the following chapters:

- Chapter 2, "Facility Description"
- Chapter 4, "Safety Structures, Systems, and Components"
- Chapter 5, "Derivation of Technical Safety Requirements"
- Chapter 6, "Prevention of Inadvertent Criticality"
- Chapter 10, "Initial Testing, In-Service Surveillance, and Maintenance"
- Chapter 11, "Operational Safety"

These chapters do not provide exhaustive listings but, instead, reference documents that indicate the major codes and standards that are applicable.

Some requirements for design features and operating practices for the specific purpose of facilitating future D&D are found in DOE and Nuclear Regulatory Commission (NRC) documents. Section 1300-11.1 of DOE Order 6430.1A (Ref. 3) requires the minimization of equipment in potentially contaminated spaces and gives examples of basic design considerations to allow for easy decontamination. NRC Regulatory Guide 8.8 (Ref. 4) identifies general design principles that decrease the potential for worker exposures. NRC Regulatory Guide 8.10 (Ref. 5) addresses operating procedures and practices that also contribute to reduced worker exposure.

Requirements for planning D&D projects and for performing actual D&D activities are more numerous. Again, however, the requirements that apply to D&D are often extracted from a document of broader scope. Tables 16.2-1 through 16.2-3 list the DOE Orders, federal environmental statutes, and South Carolina hazardous waste management regulations, respectively, that could be involved in D&D activities.

The Standards/Requirements Identification Document (S/RID) (Ref. 6) identifies requirements for planning and performance of D&D in Functional Area 15.0, "Decontamination and Decommissioning." The S/RID lists D&D requirements that apply to any nuclear/radiological facility whose mission is D&D. Since the RBOF and RRF are still operating, the S/RID effectively identifies the D&D requirements that will apply when the facility's mission changes to D&D.

The S/RID includes only requirements for programs, plans, and activities that are necessary and sufficient to protect the environment and the safety and health of the public and workers during D&D activities (i.e., the S/RID contains only those requirements that apply to the safety basis of the facility). A more comprehensive site document, WSRC-IM-95-05, (Ref. 7) includes all identifiable requirements, not only those relating to the safety basis. However, WSRC-IM-95-05 does not relate requirements to specific facilities or categories of facilities.

16.3 DESCRIPTION OF CONCEPTUAL PLANS

16.3.1 DECOMMISSIONING POLICY AND PROGRAM OUTLINES

The objective of D&D is to safely dispose of surplus contaminated facilities as part of the overall effort to achieve environmental restoration at the Savannah River Site (SRS). The removal of hazards, rather than simply restricting access to them, decreases residual risks following facility operation. The level of effort of D&D activities and the extent of pre-planned provisions for later D&D activities are both commensurate with the magnitude of the hazards presented by the facility and its complexity.

D&D is one of the last steps of the excess facility transition process. Decommissioning activities include, but are not limited to, the following (Ref. 8):

- Identification of surplus facilities
- Development of a facility transition plan
- Characterization of surplus facilities
- Deinventory, deactivation, and preliminary decontamination of the facilities
- Surveillance and maintenance of surplus facilities
- Investigation and recommendation of facility reuse options
- Evaluation of D&D options that consider land use, waste disposal, costs, and risks
- Preparation of a decommissioning plan
- Development and implementation of management controls, including requirements documents and implementing procedures
- Final decontamination, dismantlement, and demolition of designated surplus facilities
- Allocation/reallocation of resources, as necessary
- Proper retirement of all government equipment and materials
- Closeout of the facility-specific decommissioning project

16.3.2 DECOMMISSIONING ORGANIZATION AND RESPONSIBILITIES

Landlord responsibility for the RBOF and RRF, along with most of SRS, was transferred from DOE-Defense Programs (DOE-DP) to DOE-Environmental Management (DOE-EM) on January 23, 1995. Within DOE-EM, the facility is the responsibility of EM-60, the Office for Nuclear Material and Facility Stabilization, until D&D is begun. At that point, the facility will become the responsibility of EM-40, the Office of Environmental Restoration.

The facility is under the immediate management control of the Spent Fuel Storage Division (SFSD) of Westinghouse Savannah River Company (WSRC). SFSD is responsible for managing the facility during operations until all stored fuels are dispositioned. SFSD will also be responsible for overseeing actual D&D activities. Chapter 17 provides a more detailed description of the organizational structure for SFSD management.

16.3.3 SRS TRANSITION AND D&D PROGRAM PLAN

The SRS Transition Program supports the transition of facilities from the operating state to their eventual disposition (i.e., either reuse or D&D). With almost all of SRS having been transferred from DOE-DP to DOE-EM in January 1995, the scope of transition no longer includes preparation for this transfer.

SWE-TDD-94025 (Ref. 9) describes the general transition process for SRS. The transition process can result in either reuse of the facility for something other than the original purpose or D&D. The current mission of the RBOF and RRF continues to be its original purpose, the storage of fuel; therefore, no actual D&D activities have yet been considered or planned. If D&D is the selected end state for a facility, the additional management controls described in Section 16.3.4 are applicable.

16.3.4 D&D PROGRAM OVERVIEW

DOE Order 5820.2A (Ref. 10), the EM-40 Guidance Document for D&D (Ref. 11), and the EM-60 Guidance Document for Deactivation (Ref. 12) require development of a structured program for the management of any D&D project. A D&D program for dispositioning a surplus facility is approached in phases as follows:

- Phase I Transition
- Phase II Project Preparation
- Phase III Environmental Review
- Phase IV Engineering and Planning
- Phase V Decommissioning Operations
- Phase VI Closeout and Verification
- Phase VII Post-Operations Activities

While many activities of these phases are to be documented in specific reports and plans, three more comprehensive plans are required: a project management plan, a deactivation project plan, and a decommissioning plan.

The project management plan is a general document that sets forth the plans, organization, and systems for managing the project. Among the support activities covered are the following:

- D&D requirements development
- Program planning and management
- Management organization and responsibility
- Environmental, safety, and health compliance
- Waste management
- Property management
- Quality assurance
- Safety and health
- Training
- Contract administration

The deactivation project plan focuses on actions needed to reach a long-term safe storage condition prior to the start of decommissioning. The decommissioning plan, written in Phase IV, addresses the detailed steps in demolishing the facility.

The SRS D&D Program Management Plan (Ref. 13) addresses the management of site D&D projects generically and provides more detailed information on the D&D process at SRS.

16.3.5 FACILITY DESIGN FEATURES AFFECTING D&D

For a facility that is already complete, a SAR describes only the design features provided to facilitate D&D that were incorporated during major modifications (Ref. 2). In the absence of any major modifications for the RBOF and RRF, this section presents examples of original design features that could facilitate D&D.

16.3.5.1 Original Design Features

DOE Order 5820.2A (Ref. 10) requires that facilities be designed to simplify D&D and/or to increase the potential for reuse of the facility. DOE Order 6430.1A (Ref. 3) establishes the design features required in the construction of nuclear facilities to facilitate D&D activities. Since the design of the RBOF and RRF preceded these requirements by many years, design records do not indicate ease of D&D as the sole purpose of any design features. However, many features provided for operational considerations also facilitate D&D. To take best advantage of these design features during the D&D process, careful planning of the order of D&D activities is necessary. The following paragraphs provide several examples of original design features, as indicated in Chapter 2, that could facilitate D&D.

The RBOF is designed such that it could recover from mishaps involving the release of as much as 1,000,000 Ci of radioactive contaminants to the fuel handling and storage system. Water

cleanup systems can capture contaminants, thus enabling continued facility use and facilitating removal of the contaminants. The location of the RBOF and RRF (adjacent to a waste storage facility) also facilitates the treatment and storage of the contaminants. The water cleanup systems could be used during D&D to enable underwater cleaning of contaminated equipment or of the surfaces of the basins and transfer canals themselves.

Underwater cask unloading and fuel element handling was selected as the primary shielding method based on experience with similar facilities. Should shielded storage of equipment or debris be needed during D&D, the same method would be available.

The facility design provides shielding at various locations to protect personnel from direct gamma radiation. Reinforced-concrete walls that surround the waste cell, decontamination cell, and resin regeneration cell protect personnel from radiation emanating from these cells. Concrete shielding is used where required to reduce personnel radiation exposure during routine operation of the basin water handling system. The higher radiation areas have thicker walls and shielded entryways to help reduce exposure. Contaminated water is disposed of by the use of imbedded or shielded pipes, valves, pumps, and tanks. During D&D of adjacent spaces, the same shielding will protect the workers from the higher radiation in the cells and basins.

Personnel within the RBOF Control Room are shielded from radiation emanating from the basins by a 1-foot-thick concrete shield wall that permits operation of the facility even in the event of a significant release of radioactivity within the basin area. A shield wall in the RRF Control Room protects personnel from radiation emanating from the resin regeneration cell and the portable deionizers positioned adjacent to the hose box station. Each of these shielded positions could serve as a safe control station for remote D&D activities.

Fuel handling and/or storage basins and associated transfer canals serve as secondary containment (primary containment is provided by fuel cladding) for radioactive contaminants contained in the basin water. The concrete basin walls are 2-3 feet thick at grade level, to 6.5 feet at 29 feet below grade level, and the concrete basin floors are 3 feet thick. The basins are filled with deionized water. All basin floors except the Main Storage Basin have a 0.25-inch stainless steel lining. The Main Storage Basin has a 0.125-inch stainless steel lining. The walls of the disassembly, inspection and repackaging basins are lined with 0.125-inch stainless steel. The other basins have an epoxy (Phenoline AE-600[®]) coating. This protective coating prevents leaching of the concrete and provides a smooth surface that is easily decontaminated. A canal system connecting the basins is divided by an underwater door to provide a buffer zone between the disassembly basin and the major part of the facility. All portions of the canal system are 3 feet wide and 29 feet deep. The floor and walls of the canal system are covered with "Phenoline" AE-600. The canal will be easier to decontaminate and decommission because of this covering on the floor and walls.

Basin water levels are maintained above the submerged fuel to effectively reduce the exposure to personnel from gamma radiation. Chapter 2 indicates that alarms and interlocks are provided to prevent lowering the basin water level below a safe point, and the system is interlocked such that a high water level in any basin closes the inlet valve for that basin. Should the basins/canals be

used to shield any products of the D&D effort, the same water level assurance measures will ensure adequate shielding of personnel.

The RBOF is designed such that the consequence of most equipment failures is only an operating delay. Loss of power merely stops work in progress because most of the system is normally static. For this reason, the standby generator is designed only to operate health monitoring systems, resulting in less electrical equipment and backup/transfer systems requiring D&D.

The RBOF detection and monitoring equipment consists of nuclear incident monitors, gamma monitors, continuous air monitors, and an assortment of manually operated detectors. Portal monitors are available at the exits of radiological areas. With the exception of nuclear incident monitors (because there will be no risk of criticality during D&D), the same equipment could be available to monitor D&D activities.

Control room annunciation is provided for abnormal water levels. Additionally, when the facility is placed into shutdown, all operational alarms (except continuous air monitors and gamma monitors, which alarm locally) are received as one Building 244-H general alarm in the Building 221-H Control Room. As long as these systems remain operable, they could be useful to report any abnormal events during D&D.

The walls of the basin area are insulated and covered on the interior surface with a layer of transite to prevent condensation within the insulation, thereby preventing contamination within the insulation and facilitating D&D.

The ventilation system serves to control the airborne radioactivity levels inside and outside the facility. The RBOF ventilation system is designed with a multiple-air-zone concept. Under this concept, ventilation air flows from areas with a low potential for contamination to areas with a higher potential for contamination. The once-through airflow system passes air through filters on the building exhaust. Building 244-H has four supply systems each supplying separate sections of the facility, while Building 245-H maintains one supply system. Air is exhausted from Building 244-H and Building 245-H through three exhaust systems servicing separate sections of the facilities. The multiple-air-zone concept decreases the spread of contamination during operation and, if it is maintained during D&D, will control the flow of air during D&D activities.

The RBOF building is maintained at less than atmospheric pressure so that all air exiting the building is filtered before being released. The basin area is kept at a lower pressure than the rest of the building, and a separate ventilation system supplies the control room, offices, and change room. After filtration, all air is discharged through a stack 5 feet in diameter and 53 feet high. The exhaust system for the process vessels in the waste cell and decontamination cell is similar to the building system but is separate and employs acid-resistant components. This system exhausts through a pipe 10 inches in diameter and 53 feet high. Filtering the exhaust prevents contaminants from being deposited on external surfaces of the facility and precludes their cleanup during D&D. Chapter 2 provides additional information concerning RBOF/RRF ventilation systems.

Type 304 commercial-grade stainless steel is used for process piping throughout the facility to ensure the prolonged life of the imbedded piping system. Water discharges from the basins through imbedded and shielded piping. The RBOF floor drain system is stainless steel pipe. The wall thickness of buried piping minimizes leakage due to piping failures, which in turn limits the spread of contamination and facilitates D&D.

A breathing air system is supplied by a 100-cfm compressor, which in turn supplies 10 outlets to which plastic suits can be connected. The reserve capacity of the air system is 5 minutes for five men. The same system could be used during D&D activities.

The installed lifting capacity of the RBOF consists of the following:

- A 100-ton bridge crane with two 50-ton hoist trolleys and two 3-ton hoists suspended on monorails
- A traveling bridge with two 3-ton monorails (one on each side of the platform)
- Five turntables
- A 3-ton motorized bridge
- Two 6,000-pound trolleys and hoists
- A 6,000-pound manually operated bridge
- 10 four-wheel trolleys with 1,000-pound chainfalls

With proper planning, existing lifting systems could be available for D&D, and their use for dismantling the facility could be maximized.

In the original design, only hand-applied fire protection methods were considered necessary; no installed sprinklers or fire hydrants were provided. The absence of installed piping, valving, sprinkler heads, and reservoirs for extinguishing agents will facilitate D&D efforts.

16.3.5.2 Modification Design Features

As indicated in Chapter 2, no major modifications to the RBOF or RRF have occurred since initial operation. However, the following upgrades or minor changes to the facility have been made:

- In 1978, the second cask basin was changed into a storage basin by adding racks.
- Heating, Ventilation, and Air Conditioning (HVAC) units and heating coils (except for reheaters) were replaced and a new chiller system was installed in 1994. These modifications made the HVAC systems more efficient by providing better flow rates and by reducing the gaseous and particulate airborne effluents.
- A new waste pump was installed in waste tank EP38 in 1983.

- The sodium nitrate tank was reactivated and converted from using sodium nitrate to using sodium nitrite. This change was made to improve chemistry control (i.e., the sodium nitrite is less caustic than the sodium nitrate).
- Two waste storage tanks, WH-1 and WH-2, were removed.
- The drains for the chemical pads have been rerouted from the 200-H-Area seepage basin to the waste tank.
- Targets irradiated in the reactor areas are no longer being cleaned in the RRF.
- In 1996, structural upgrades were made to the Building 244-H superstructure to increase the resistance to Performance Category 3 wind and seismic loads as documented in Project #S-6032 (Ref. 14).

16.3.6 DESCRIPTION OF PROGRAMMATIC D&D PROVISIONS AND ACTIVITIES

This section describes the operating practices and design features that allow for the management of hazardous materials and the programs and practices necessary to ensure that the health and safety of the public, workers, and the environment are maintained. WSRC Procedure Manual 1B (Ref. 15) provides administrative guidance for D&D of contaminated facilities. This section also discusses the health protection programs that will be in place for facility surveys and worker safety during D&D.

An important aspect of planning for facility D&D includes a review of unusual operating occurrences or accidents. Chapter 3 presents the hazards analysis for the RBOF and RRF. Data covering the period 1983 to present show that no incidents have occurred that fall outside the SAR safety envelope. Any unusual occurrences are documented in a facility occurrence database and the 200-Area Fault Tree Data Bank.

16.3.6.1 Inventory of Hazardous/Contaminated Materials

Chemicals found within the facility are inventoried on a regular basis. The hazards analysis includes a comprehensive inventory of chemical and radioactive materials. Periodic reviews ensure new chemicals that have been introduced or chemicals that have been removed are reflected in the hazards analysis.

Most dry chemicals are stored in a separate facility (Building 244-1H) where they can easily be surveyed, monitored, and controlled. Only two dry chemicals are regularly stored in Building 244-H: approximately 100 pounds each of soda ash and steam cleaner. These chemicals are stored in the manufacturer's original packaging/containers to minimize contamination/spreading of these materials and to allow for easy inventory.

As indicated in Chapter 2, liquid chemicals are stored in discrete tanks with known capacities that are clearly identified and can easily be measured, sampled, or drained. Chemical storage

tanks have gages that allow for measurement of chemical quantities. Chemicals that could result in a hazard due to mixing are segregated on separate storage pads when possible.

Radiological hazards for the RBOF and RRF are primarily confined to materials stored within the basins. Strict procedures are maintained for the handling of radioactive materials to ensure that radioactive contamination outside the basins does not occur. Regular monitoring of the facility for radiological contamination is performed by the Radiological Control and Health Physics (RC&HP) Department. Radiological surveys are performed in accordance with WSRC Procedure Manual 5Q1.2 (Ref. 16), and facility contamination levels are recorded. Locations within the building are classified in accordance with WSRC Procedure Manual 5Q (Ref. 17) based upon the expected levels of radiation or contamination. Contaminated areas are clearly identified. Section 4.8.1 provides more information on this topic.

All effluents from the facility are monitored for radioactive material content. Airborne effluents from Building 244-H are filtered through building exhaust filters and are continuously monitored. The effluents are also sampled daily before being released through the stack.

16.3.6.2 Processing, Transportation, and Disposal of Materials

SRS was acquired by the U.S. Government in 1950 and, since that time, has contracted for the design, development, construction, and operation of various facilities at SRS to support national defense and space exploration. Environmental restoration and D&D activities will generate increasing quantities of hazardous and low-level radioactive wastes. All radioactive/nonradioactive, hazardous/nonhazardous, and mixed wastes generated during the D&D processes will be managed appropriately to minimize the risk to personnel and the environment during processing, packaging, and final dispositioning of the waste materials to Treatment, Storage, or Disposal (TSD) facilities.

As noted in Chapter 9, the RBOF/RRF Waste Management Program includes both operating procedures and a waste minimization plan to ensure that the wastes generated at or leaving the RBOF/RRF site meet the applicable waste acceptance criteria of TSD facilities. As stated in the Proposed Site Treatment Plan (Ref. 18), WSRC will use onsite TSD facilities for the future dispositioning of all types of hazardous and nonhazardous wastes, including those that will be generated by the RBOF and RRF during D&D activities. These TSD facilities, some existing and some planned, form the framework for managing and subsequently reducing SRS's present and future inventory of hazardous and nonhazardous wastes.

During D&D activities, a number of wastes and probable waste sources will contribute to the overall waste management inventory at the RBOF and RRF. These wastes and probable waste sources (shown in parentheses) include the following:

- Nonradioactive solid waste (e.g., paper waste such as cups, towels, bags, and boxes; standard garbage including cleaning-type trash, metal, cans, and plastic bags; and any other waste determined by Solid Waste Management [SWM] to be "clean")

- Low-level radioactive solid waste (e.g., soil and rubble from D&D activities; spent resin beds from deionizers; scrap metal; and job control wastes such as filters, tools, plastic suits, rubber gloves, and spray cans)
- Intermediate-level radioactive solid waste (e.g., scrap metals from the RBOF basin and used canyon and tank farm equipment contaminated with fission products and/or induced activity)
- Liquid waste (e.g., potential residues in system equipment piping from RBOF and RRF operations such as filter flushes, deionizer operations, and cask/fuel operations)
- Gaseous waste (e.g., gaseous radionuclides and airborne radioactive particulates)
- Radioactive mixed waste (none expected)
- Hazardous chemical waste (e.g., the chemicals sodium hydroxide, nitric acid, and sodium nitrite; and chemical residues collected in waste tank EP38)

Standard site waste management terminology classifies solid low-level waste with radiation levels below 200 mrem/hr at 5 cm from the exterior of an open container as low-activity waste. The low-activity waste resulting from D&D activities at the RBOF and RRF will include job control waste, scrap metal, spent resin beds from deionizers, and contaminated soil and rubble. The job control waste will consist of contaminated protective clothing including plastic suits, shoe covers, lab coats, and plastic sheeting. The scrap metal will include contaminated tools, process equipment, and laboratory equipment. The majority of soil and rubble will be generated from facility D&D, demolition, and cleanup activities.

Standard site waste management terminology classifies radioactive waste with radiation levels below 200 mrem/hr at 5 cm from the exterior of a closed waste container as intermediate-activity waste. The intermediate-activity waste resulting from D&D activities will consist primarily of such items as highly contaminated lab coats, shoe covers, and plastic sheeting. This waste is assumed to be combustible and contaminated primarily with fission products. Scrap intermediate-activity waste will consist of contaminated facility or reactor hardware, reactor fuel fittings and target fittings, jumpers, and used canyon and tank farm equipment contaminated with fission products and/or induced activity. These fission products will be both loose and fixed surface contamination. The intermediate-activity waste will be disposed of in the intermediate-level nontritium vault.

TRANSPORTATION

Although not governed by Department of Transportation (DOT) regulations, onsite transfers of hazardous and nonhazardous materials are controlled to protect the health and safety of employees and to prevent releases of undesirable materials to the environment. Onsite transfers of hazardous materials, substances, and wastes from H Area to TSD facilities are conducted on controlled roads within a fenced and guarded area. Since the general public does not have access to these roads, the material movements are considered intrafacility transfers; however, several

public access roads may be used for hazardous material transport (e.g., South Carolina state roads 125 and 1A). Use of these roads requires either the implementation of transport provisions to ensure the safety of the general public or the compliance with DOT (offsite) shipping requirements. WSRC Procedure Manual 19Q (Ref. 19) provides information on the use of SRS public access roads.

Site policy requires that onsite intrafacility transfers of hazardous materials by commercial carrier or public vehicle comply with the requirements of DOE, DOT, Environmental Protection Agency, and NRC standards and follow the site transportation guidelines detailed in Manual 19Q. Manual 19Q demonstrates compliance with DOE mandates and provides the administrative controls and procedural requirements for conducting transportation activities in a safe manner. These administrative controls and procedural requirements include, but are not limited to, the following:

- All onsite transfers and/or offsite shipments of hazardous materials, substances, and wastes
- Hazards classification
- Packaging
- Marking and labeling
- Placarding
- Loading and unloading
- Transporting and in-transit storage

Manual 19Q does not apply to the movement of hazardous materials within a facility. Movements within a facility are governed by Occupational Safety and Health Administration standards and are addressed in applicable site documents.

The exteriors of packages moved between site areas are marked or labeled to identify the material and its associated hazards. Containers of radioactive material are marked or labeled "Radioactive." Movements of accountable Special Nuclear Material (SNM) are coordinated with the site Material Control and Accountability (MC&A) Section and the facility Material Balance Area Custodian in accordance with applicable MC&A procedures, including WSRC Procedure Manual 14Q and the MC&A Plan (Ref. 20, 21).

As required by applicable safeguards and security DOE Orders, packages determined to contain classified material or SNM are sealed with tamper-indicating devices for movements of material between protected areas at SRS, or between protected areas and staging areas, and are under direct surveillance by an appropriate number of security personnel necessary to protect against threats established by site policy.

Hazardous materials transported in commercial vehicles are properly classified, packaged, marked, labeled, placarded, and described on shipping papers as required by DOT regulations

(Ref. 22), DOE Orders 5480.3 and 1540.2 (Ref. 23, 24), and Manual 19Q. Domestic offsite shipments of Category I SNM are made by the DOE Transportation Safeguards System under the auspices of the DOE's Albuquerque Operations Office.

Waste Packaging

When shipping a hazardous material, DOT-approved packaging is used as listed in Title 49 Code of Federal Regulations (CFR) Part 172.101 (Hazardous Materials Table, Columns a and b) (Ref. 22). This packaging includes industrial packaging, Type A and B packaging (which are approved by DOT and listed in 49 CFR 173.415 and 173.416, respectively), and DOT specification packaging. This packaging, when used as authorized, requires no further review or regulatory approval. Each type of packaging is briefly described as follows:

- Industrial packaging, otherwise described in the regulations as "Strong, Leak-Tight Packaging," meets approved industry standards for the containment of any material.
- Type A packaging is designed to retain its integrity and shielding attributes under normal conditions of transportation. The packaging must pass tests prescribed for maintaining packaging integrity. Type A packaging is used to package "A₁" or "A₂" quantities of radioactive material as limited in 49 CFR 173.433 and /or the "Table of A₁ or A₂ Values for Radionuclides" in 49 CFR 173.435.
- Type B packaging is used to ship quantities of radioactive material that exceed the amount specified for "A₁" or "A₂" quantities in Type A packaging. Type B packaging is designed to withstand normal conditions of transportation, including hypothetical accident test conditions set forth in 10 CFR 71 (Ref. 25). Authorized Type B packaging is listed in 49 CFR 173.416.
- DOT specification packaging, as listed in 49 CFR 178, may be purchased and/or fabricated according to the listed requirements. DOT specification packaging is approved for use in accordance with the specific requirements for the materials being shipped.

The use of approved packaging for shipments of material in quantities or forms not previously approved requires an "exemption" as described in 10 CFR 71, Subpart B and/or 49 CFR 173.3. Approval for an "exemption," a new packaging design, or a change in the use of existing packaging is based upon the SAR for the particular type of packaging, prepared to demonstrate the safety of the packaging and its intended use.

All waste disposal containers are required to be engineered concrete or metal containers that have been approved by SWM (Ref. 26). Waste management procedures are in place that define this approval process and require that SWM Engineering, SWM Operations, and SWM Maintenance concur that the container can be safely handled, does not impair vault space use, and satisfactorily contains the waste contents.

The majority of low-level radioactive solid wastes are containerized for permanent disposal in standard 1.2- by 1.2- by 1.8-meter metal containers (B-25 boxes), 0.6- by 1.2- by 1.8-meter metal containers (B-12 boxes), or DOT-approved 210-liter metal drums. These containers have been approved for use at the newly constructed low-activity waste vaults in E Area. Other containers are specific to the generator of the waste form and are approved for either one-time use or unlimited use, depending on the disposal circumstances.

WASTE TREATMENT FACILITIES

Presently, certain federal and state regulations restrict the in-ground disposal of hazardous or mixed hazardous waste without prior treatment to remove or stabilize the hazardous constituent. A number of onsite facilities satisfy this treatment requirement, including the following:

- The Consolidated Incineration Facility incinerates a number of low-level radioactive solid and liquid waste streams, converting the material to ash, which is then stabilized in grout or concrete for disposal.
- The F-Area and H-Area Effluent Treatment Facility uses ion exchange to remove hazardous constituents from certain liquid waste streams.
- Several new or planned facilities use macroencapsulation or vitrification as the preferred treatment technology.
- Vendor or other DOE technologies and/or expertise are used for the treatment of certain mixed waste streams (e.g., lead) by decontamination, amalgamation, stabilization, or treatment by wet oxidation.

As noted in the Proposed Site Treatment Plan (Ref. 18), the choice of a final treatment option for a certain waste stream is subject to change. Although a preferred treatment option is described, evolving technologies and/or future DOE funding considerations may result in changes to these results.

WASTE DISPOSAL FACILITIES

Low-level, nonhazardous radioactive wastes are packaged in approved containers and disposed of in the new E-Area vaults, a 200-acre facility immediately north of the current low-level waste burial site. The E-Area vaults contain several large concrete vaults divided into cells and consist of three types of structures to house four designated waste types, including the following:

- Low-activity waste vaults for wastes with radiation levels below 200 mrem/hr at 5 cm from the exterior of the waste container or less than 10 Ci of tritium per package
- Long-Lived Waste Storage Building for long-term storage of waste containing long-lived isotopes that exceed disposal facility performance criteria

- Intermediate-level tritium vaults and intermediate-level nontritium vaults that are designed for tritium and nontritium contaminated waste, respectively, with radiation levels greater than 200 mrem/hr at 5 cm from the exterior of the waste container

Additionally, plans are to construct trenches in E Area for the disposal of suspect soil (i.e., soil from regulated areas designated as potentially contaminated) and for the disposal of containers containing naval reactor components (e.g., control rods, control rod drive mechanisms, resin vessels, adapter flanges, and similar equipment).

Treated hazardous and mixed wastes are transported to the Hazardous Waste / Mixed Waste Disposal Facility located near the northeast corner of the E-Area vaults disposal site. This facility provides a disposal facility permitted in accordance with the Resource Conservation and Recovery Act for treated hazardous and mixed wastes that cannot be disposed of in existing planned facilities at SRS.

16.3.6.3 Radioactive and Hazardous Waste Management

The guidelines of DOE and other federal and state regulatory agencies require that hazardous (including radioactive) wastes be managed, treated, stored, and disposed of in a manner that protects the public health and safety, the environment, and groundwater resources. During D&D operations, the level of radioactivity in any effluent released to the environment is maintained as low as reasonably achievable, and waste management costs versus economic benefit are evaluated and maintained as low as economically achievable. Primary waste management policies and good practices in effect during RBOF and RRF operation would also be in effect during D&D, such as the following:

- Controlling the overall quantity of waste material produced
- Characterizing, classifying, certifying, and safely segregating hazardous wastes at the facility
- Managing hazardous waste releases

As noted in Chapter 9, waste management activities at the RBOF and RRF are consistent with national, state, and procedural requirements, including DOE Orders 5820.2A and 5400.1 (Ref. 10, 27). DOE Order 5820.2A establishes policies, guidelines, and minimum requirements for the management of radioactive waste, mixed waste, and contaminated facilities at DOE sites. This Order also addresses the storage, treatment, and disposal of high-level waste, mixed waste, low-level waste, transuranic waste, and naturally occurring and accelerator-produced radioactive materials that are generated by DOE operations. Low-level waste is defined in the Order as waste that contains radioactivity and that is not classified as high-level waste (i.e., waste material that results from the reprocessing of spent nuclear fuel), transuranic waste, spent fuel, or byproduct material. In addition, Chapter III of the Order requires the DOE field sites to prepare and maintain a site-specific radiological performance assessment for any low-level waste disposal facility located at DOE field sites. A radiological performance assessment provides

reasonable assurance that the facility design and method of disposal comply with the performance objectives of the Order.

Chapter 9 provides more information on waste management.

16.3.6.4 Planned Final Radiation Survey

WSRC Procedure Manual 5Q1.2 (Ref. 16) prescribes the method and frequency of radiological contamination surveys. Surveys will be performed following facility shutdown to ensure the safety of workers and the environment during any safe storage period that may precede D&D. Surveys will continue during D&D to monitor contamination migration. A final contamination survey will be conducted to verify that predetermined criteria for either free release or restricted use, depending on the selected end state of the facility, have been achieved.

16.3.6.5 Methods for Protecting Worker and Environment from Radiation Hazards

Chapter 7 of the Generic SAR (Ref. 28) describes the Radiological Protection Program for workers. During D&D of the RBOF and RRF, contamination will be detected through the same survey program used during operation. WSRC Procedure Manual 5Q1.2 (Ref. 16) controls the survey program.

16.3.6.6 Description of Radiological Protection Program to be in Effect during Decommissioning

During D&D of the RBOF and RRF, the RC&HP Department will continue to be responsible for radiological protection of the site personnel and the public as they were during operation. Chapter 7 of the Generic SAR (Ref. 28) describes the Radiological Protection Program.

While the activities that are controlled under the Radiological Protection Program are generally different from the activities that will be performed during D&D, the same basic program is equally applicable.

16.3.6.7 Safety Analysis of D&D Activities

The current mission of the RBOF and RRF continues to be its original purpose, the storage of fuel. No actual D&D activities have yet been considered or planned. As the specific D&D project plan for the RBOF and RRF is prepared, a systematic method for safety analysis of D&D activities will be developed in order to preserve the safety basis of the facility.

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Aiken, SC, March 1995.

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16.5 TABLES

Table 16.2-1 DOE Orders Applicable to D&D Operations^a

Order	Description
1540.2	Hazardous Material Packaging for Transport - Administrative Procedures
1540.3	Base Technology for Radioactive Material Transportation
5400.1	General Environmental Protection Program
5400.2A	Environmental Compliance Issue Coordination
5400.3	Hazardous and Mixed Waste Program
5400.5	Radiation Protection of Public and the Environment
5440.1E	National Environmental Policy Act Compliance Program
5480.1B	Environment, Safety, and Health Program for DOE Operations
5480.3	Safety Requirements for the Packaging and Transportation of Hazardous Materials
5480.4	Environmental Protection, Safety, and Health Protection Standards
5480.11	Radiological Protection for Occupational Workers
5480.21	Unreviewed Safety Questions
5480.22	Technical Safety Requirements
5480.24	Nuclear Criticality Safety (being replaced by DOE Order 420.1)
5483.1A	Occupational Safety and Health Program for DOE Contractor Employees at Government-Owned Contractor-Operated Facilities
5484.1	Environmental Protection Safety and Health Protection Information Reporting
5820.2A	Radioactive Waste Management

^a Note: This list is not all-inclusive.

Table 16.2-2 Federal Environmental Statutes

Act	Purpose
Clean Air Act	To protect and enhance the quality of the nation's air and to support a national research and development program to prevent and control air pollution
Clean Water Act	To restore and maintain the integrity of the nation's waters including both surface and ground waters
Marine Protection, Research, and Sanctuaries Act	To regulate the transportation of material for the purpose of dumping the material into ocean waters
Noise Control Act	To establish federal noise emission standards for products distributed in commerce
Occupational Safety and Health Act	To authorize the establishment of mandatory occupational safety and health standards and to provide medical criteria that ensure, insofar as practicable, that no employee will suffer diminished health, functional capacity, or life expectancy as a result of his work experience
Resource Conservation and Recovery Act	To promote the protection of health and the environment to conserve valuable material and energy resources by ensuring that hazardous waste management practices are conducted and by requiring that hazardous waste be properly managed in the first instance, thereby reducing the need for corrective action at a future date
Safe Drinking Water Act	To provide contamination criteria for public water systems that have at least 15 service connections or regularly serve at least 25 individuals
Comprehensive Environmental Response, Compensation and Liability Act	To provide cleanup standards, the establishment of a fund, and civil penalties and awards in an effort to ensure the reclamation of contaminated private and government-owned sites
National Environmental Policy Act of 1969	To determine whether the activity performed will have an adverse effect on the environment

Table 16.2-3 South Carolina Hazardous Waste Management Regulations

Act	Purpose
124	General Program Requirements
260	Hazardous Waste Management System: General
261	Identification and Listing of Hazardous Waste
262	Standards Applicable to Generators of Hazardous Waste
263	Standards Applicable to Transporters of Hazardous Waste
264	Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities
265	Interim Status Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities
266	Standards for the Management of Specific Hazardous Wastes and Specific Types of Hazardous Waste Management Facilities
268	Land Disposal Restrictions
270	Permit Requirements

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

CHAPTER 17

MANAGEMENT, ORGANIZATION, AND INSTITUTIONAL SAFETY PROVISIONS

AUGUST 1997

UNCLASSIFIED DOES NOT CONTAIN UNCLASSIFIED CONTROLLED NUCLEAR INFORMATION	
ADC & Reviewing Official:	<u>J.N. Dewes</u>
J.N. DEWES, Manager, Regulatory Programs	
Date:	<u>9/23/97</u>

Westinghouse Savannah River Company
Projects, Engineering & Construction Division
Aiken, SC 29808

SAVANNAH RIVER SITE

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RECEIVING BASIN FOR OFFSITE FUELS AND THE RESIN REGENERATION FACILITY SAFETY ANALYSIS REPORT

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SAVANNAH RIVER SITE

PREPARED FOR THE U.S. DEPARTMENT OF ENERGY UNDER CONTRACT NO. DE-AC09-96SR18500

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ACRONYMS

DOE	Department of Energy
ESH&QA	Environmental, Safety, Health, and Quality Assurance
MP	Management Policy
MRP	Management Requirement and Procedure
QA	Quality Assurance
PECD	Projects, Engineering, and Construction Division
RBOF	Receiving Basin for Offsite Fuels
RBOSC	RBOF/RRF Operations Safety Committee
RDP	Reactor Division Procedure
RRF	Resin Regeneration Facility
SAR	Safety Analysis Report
SFSD	Spent Fuel Storage Division
SRS	Savannah River Site
SRTC	Savannah River Technology Center
USQ	Unreviewed Safety Question
USQD	Unreviewed Safety Question Determination
WSRC	Westinghouse Savannah River Company

17.0 MANAGEMENT, ORGANIZATION, AND INSTITUTIONAL SAFETY PROVISIONS

17.1 INTRODUCTION

17.1.1 OBJECTIVE

Chapter 17 of the Generic Safety Analysis Report (SAR) (Ref. 1) describes the overall safety management policies and programs for the Savannah River Site (SRS). The purpose of this chapter is to provide information on the safety management policies and programs that pertain specifically to the Receiving Basin for Offsite Fuels (RBOF) and the Resin Regeneration Facility (RRF). Information is provided to demonstrate that the RBOF/RRF management organization is structured to support safe facility operation and that the safety management programs used by the facility promote a safety-conscious environment to ensure that hazards and safety issues are identified, communicated, evaluated, resolved, and documented.

17.1.2 SCOPE

This chapter presents information on the following:

- Management, technical, and other site organizations that support safe facility operation
- Responsibilities of, and relationships between, non-operations organizations having a safety function
- Interfaces of the operations and non-operations organizations
- Requirements for safety management programs that promote safety consciousness and morale, including safety review and performance assessment, configuration and document control, occurrence reporting, and safety culture
- Safety management policies and programs to demonstrate that facility operations are embedded in a safety-conscious environment

Chapter 17 of the Generic SAR (Ref. 1) describes the overall site program. This chapter describes the features of the program that are specific to the RBOF and RRF or that are different from the program described in the Generic SAR.

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17.2 REQUIREMENTS

Section 17.2 of the Generic SAR (Ref. 1) lists the applicable Department of Energy (DOE) Orders and the associated implementing site-level management, organizational, and institutional safety program documents.

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17.3 ORGANIZATIONAL STRUCTURE, RESPONSIBILITIES, AND INTERFACES

17.3.1 ORGANIZATIONAL STRUCTURE

In 1994, responsibility for operation of the RBOF and RRF was transferred from the Nuclear Materials Stabilization Program Division to the Spent Fuel Storage Division (SFSD).

Figure 17.3-1 illustrates the basic organizational structure of the RBOF and RRF.

17.3.2 ORGANIZATIONAL RESPONSIBILITIES

Chapter 17 of the Generic SAR (Ref. 1) provides the Westinghouse Savannah River Company (WSRC) overall organizational responsibilities in terms of the interrelationships with other WSRC divisions. The Reactor Areas Manager is responsible for the overall safety and safe operation of SFSD facilities, which include the RBOF and RRF. The Basin Facility Manager is responsible for the overall safety and safe operation of the RBOF and RRF. Additional managers are responsible for providing engineering, Quality Assurance (QA), regulatory, training and procedure support to the RBOF and RRF.

17.3.2.1 Management Staff

Section 17.3.2.1 of the Generic SAR (Ref. 1) describes the overall responsibilities of the WSRC management staff. Specific responsibilities of SFSD managers are as follows (Ref. 2):

REACTOR AREAS MANAGER

The Reactor Areas Manager has the following responsibilities:

- Operating the reactor areas facilities, including the RBOF and RRF, in accordance with the authorization basis
- Maintaining the facilities in safe conditions at all times
- Controlling special nuclear material
- Ensuring that the operations staff is trained, qualified, and certified to safely operate and maintain the facilities in a safe and efficient manner
- Coordinating plant security and access control requirements for the facilities
- Coordinating all SFSD waste management activities
- Developing and implementing initial and continuing training programs for operations, maintenance, and technical support personnel

- Developing and controlling operations, maintenance, and division administrative procedures
- Administering and monitoring the qualification programs for operations personnel
- Coordinating work group activities including recertification, maintenance, construction, and technical organizations through established work control procedures
- Performing event assessments and monitoring the corrective action process to ensure that corrective measures are established and implemented properly
- Certifying operations personnel
- Supporting and performing appropriate portions of the division's self-assessment process; providing assurance that SFSD organizations are meeting requirements; and identifying improvements to enhance performance effectiveness

BASIN FACILITY MANAGER

The Basin Facility Manager has the following responsibilities:

- Operating the RBOF and RRF in accordance with the approved authorization basis
- Facilitating and controlling procedure changes and physical changes in plant configuration
- Coordinating the activities of all work groups within the RBOF and RRF
- Ensuring that at least one Shift Supervisor is present in the facility during operation and that at least one qualified Shift Supervisor or operator is either present in the facility or manning a remote monitoring station in Building 221-H while in shutdown
- Ensuring that on-call support personnel are assigned and that technical support personnel are available to provide technical assistance to the production staff
- Ensuring that facility control is carried out by qualified operators in accordance with written procedures
- Ensuring that RBOF/RRF Shift Supervisors and operators are subject to limitations when assigned work outside of their regular schedules
- Coordinating with line and support organizations to develop and maintain the RBOF/RRF operating schedule
- Selecting, training, and qualifying operations personnel
- Implementing RBOF/RRF performance goals and standards
- Conducting investigations of abnormal operational events to identify and correct the root cause of the event
- Providing input to the Reactor Areas Manager for the annual operating budget and ensuring that operations adhere to budgetary constraints

- Providing oversight and assessment of all operations activities
- Identifying and correcting operator performance deficiencies
- Supporting and performing appropriate portions of the division's self-assessment process; providing assurance that RBOF/RRF organizations are meeting requirements; and identifying improvements to enhance performance effectiveness

SFSD ENGINEERING MANAGER

The SFSD Engineering Manager has the following responsibilities:

- Providing engineering and technical support to SFSD
- Developing functional requirements for projects and acting as technical liaison during project implementation
- Defining division policy and procedures for design control, engineering development, configuration management, and nuclear safety processes
- Approving division-level implementing procedures for engineering and technical programs
- Managing and providing technical direction for the division's plant modification process
- Maintaining and implementing the division's Unreviewed Safety Question Determination (USQD) process in accordance with WSRC requirements
- Providing the primary interface with the Projects, Engineering, and Construction Division (PECD) in the development, issuance, and maintenance of SFSD authorization basis documents
- Coordinating environmental programs and issues to ensure compliance with WSRC and external regulations and requirements
- Coordinating DOE Order and Directive compliance activities
- Supporting and performing appropriate portions of the division's self-assessment process; providing assurance that RBOF/RRF organizations are meeting requirements; and identifying improvements to enhance performance effectiveness

FACILITIES SUPPORT MANAGER

The Facilities Support Manager has the following responsibilities:

- Providing maintenance support to SFSD organizations to maintain facilities in accordance with the division's operating plan
- Providing planning support to SFSD departments

- Coordinating the RBOF/RRF “Plan of the Day” process
- Maintaining and tracking the RBOF/RRF weekly schedule performance indicators
- Supporting and performing appropriate portions of the division’s self-assessment process; providing assurance that RBOF/RRF organizations are meeting requirements; and identifying improvements to enhance performance effectiveness

QUALITY ASSURANCE MANAGER

The Quality Assurance Manager has the following responsibilities:

- Providing QA and quality control support
- Developing and coordinating the division’s self-assessment processes to provide assurance that SFSD organizations are meeting requirements and to identify improvements to enhance performance effectiveness

17.3.2.2 Technical and Engineering Support, Maintenance, and Modifications

Engineering and support services are provided by SFSD Engineering staff if possible and by PECD, especially for large projects and such services as safety analyses. Section 17.3.2.2 of the Generic SAR (Ref. 1) describes the PECD support functions.

17.3.2.3 Safety Issue Discovery, Communication, Management, and Resolution

As discussed in Section 17.3.2.3 of the Generic SAR (Ref. 1), each WSRC employee is responsible for reporting safety-related concerns and occurrences and initiating, if necessary, a request for corrective action. Procedure 16-1 of WSRC Procedure Manual 1Q (Ref. 3) defines the requirements and responsibilities for corrective actions and is directly implemented by SFSD.

17.3.2.4 Independent Safety Review, Audit, and Compliance Determination

Oversight activities for the RBOF and RRF consist of the line self-assessments and independent assessments that make up the SRS Oversight Program. Section 17.4.1.1 describes these oversight activities for the RBOF and RRF. Chapters 14 and 17 of the Generic SAR (Ref. 4, 1) provide additional information on the SRS Oversight Program.

17.3.2.5 Safety Analysis Services

The Savannah River Technology Center (SRTC) and organizations within PECD develop new technologies associated with the SRS mission, refine existing technologies to meet evolving needs, and provide analytical and experimental technical aspects of safety, environmental, and

production support to the RBOF and RRF. Section 17.3.2.5 of the Generic SAR (Ref. 1) describes SRTC and PECD responsibilities related to safety analysis services.

17.3.2.6 Support Services

Sitewide safety-related support services are provided by Safeguards, Security and Emergency Services and the Site Services Department. Section 17.3.2.6 of the Generic SAR (Ref. 1) describes the responsibilities of these organizations.

17.3.3 STAFFING AND QUALIFICATIONS

The requirements for personnel selection, qualification, and training are specified to ensure that the facility can perform its mission in a safe and efficient manner. Figure 17.3-1 shows the organization chart for the RBOF and RRF. Section 17.3.3 of the Generic SAR (Ref. 1) provides sitewide requirements on staffing, qualifications, and training, which are also applicable to the RBOF and RRF.

Procedure RDP 2.20 of WSRC Procedure Manual RD-1 (Ref. 2) describes the staffing levels for SFSD facilities, which include the RBOF and RRF. RDPs 6.01 and 6.07 of Manual RD-1 provide division- and facility-level requirements on the qualifications and training of facility personnel.

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17.4 SAFETY MANAGEMENT POLICIES AND PROGRAMS

The key elements in the safety program are the strength and consistency of SRS management support and involvement. It is the policy of SRS management that employee safety is more important than production, quality, or costs. The commitment of SRS management to the principles of safety is demonstrated by the following:

- Management's comprehensive understanding of the site progress and problems in enhancing safety
- Implementation of administrative programs and procedures, in conjunction with area design, to ensure that safety is maintained to the highest extent practical in conformance with DOE Order 5480.23 (Ref. 5)
- Management's assignment of high priority for notification and procedural changes to enhance facility safety
- Management's review and approval of realistic and measurable safety goals to ensure that these goals reflect a full commitment to safety principles
- Monitoring of monthly progress towards the safety goals by SRS managers and supervisors
- Establishment of a training program to educate employees on safety procedures and practices and to emphasize that compliance with these procedures and practices is a condition of employment

The safety program consists of guidelines, management and worker philosophies, practices, instructions, and procedures for administrative controls of site personnel activities, work practices, and records.

17.4.1 SAFETY REVIEW AND PERFORMANCE ASSESSMENT

17.4.1.1 Oversight Activities

Oversight activities for the RBOF and RRF consist of the line self-assessments and independent assessments that make up the SRS Oversight Program. Chapters 14 and 17 of the Generic SAR (Ref. 4, 1) provide additional information concerning the SRS Oversight Program.

Line self-assessments for the RBOF and RRF are the responsibility of the Basin Facility Manager. The RBOF/RRF Operations Safety Committee (RBOSC), established in accordance with requirements in WSRC Procedure Manual RD-1 (Ref. 2), advises the Basin Facility Manager on all matters affecting the operation of the RBOF and RRF and on associated activities that affect safety. The RBOSC functions as the Facility Operations Safety Committee as discussed in Section 17.4.1 of the Generic SAR.

The independent assessments for the RBOF and RRF are performed by engineering, safety, and operations evaluation boards that are described in Section 17.4.1 of the Generic SAR.

17.4.1.2 ESH&QA Independent Oversight Responsibilities

The Environmental, Safety, Health, and Quality Assurance (ESH&QA) Division has responsibility for providing independent oversight of environmental, safety, health, safeguards, security, and QA activities, as discussed in Section 17.4.1.2 of the Generic SAR (Ref. 1), for the RBOF and RRF.

The ESH&QA Division provides independent oversight of the safety of operations of SRS facilities. Within ESH&QA, the Facility Safety Evaluation Section performs this function through an ongoing program of selective independent review, surveillance, and assessment of activities related to nuclear safety.

17.4.1.3 Unreviewed Safety Question Determinations

USQDs are performed to determine whether or not a proposed change, test, or experiment involves an Unreviewed Safety Question (USQ) as defined in DOE Order 5480.21 (Ref. 6). Section 17.4.1.3 of the Generic SAR (Ref. 1) discusses conditions that result in a USQ. The RBOF and RRF implement USQDs in accordance with WSRC Procedure Manual 11Q (Ref. 7).

17.4.1.4 Self-Assessments and Independent Assessments

Section 17.4.1.1 describes line self-assessments and independent assessments for the RBOF and RRF.

17.4.1.5 Lessons Learned Program

Section 17.4.1.5 of the Generic SAR (Ref. 1) discusses the SRS Lessons Learned Program. SFSD implements the SRS Lessons Learned Program based on internal and external operating experience in accordance with Management Policy (MP) 4.19 of WSRC-1-01 and Management Requirement and Procedure (MRP) 4.14 of WSRC Procedure Manual 1B (Ref. 8, 9).

17.4.2 CONFIGURATION AND DOCUMENT CONTROL

17.4.2.1 Configuration Management

The Configuration Management Program serves to evaluate and control changes in facility components, systems, and operating procedures. Its mission is to establish programs for a

technical baseline, change control, and document control and verification. Section 17.4.2.1 of the Generic SAR (Ref. 1) describes the SRS Configuration Management Program.

17.4.2.2 Management Requirements and Procedures for Document Control

Section 17.4.2.2 of the Generic SAR (Ref. 1) lists the MRPs for WSRC document control activities applicable to the RBOF and RRF.

17.4.2.3 Quality Assurance

Section 17.4.2.3 of the Generic SAR (Ref. 1) discusses QA of configuration management activities applicable to the RBOF and RRF.

17.4.3 OCCURRENCE REPORTING

Section 17.4.3 of the Generic SAR (Ref. 1) describes the site occurrence reporting process, which is applicable to the RBOF and RRF.

17.4.4 SAFETY CULTURE

Section 17.4.4 of the Generic SAR (Ref. 1) describes the policies and programs used to promote the interest and involvement of all associated workers in facility safety, to facilitate a questioning attitude toward safety-related activities and equipment, and to ensure that workers understand the potential risks to the facility and fellow workers, as well as the rewards and sanctions associated with personal safety performance. The Generic SAR section also describes the following WSRC MPs, which are contained in WSRC-1-01 (Ref. 8):

- MP 1.1, "Total Quality and Commitment to Excellence"
- MP 1.2, "Management Policies, Requirements, and Procedures System"
- MP 1.10, "Employee Communications"
- MP 1.11, "Open Communication of Employee Concerns"
- MP 1.18, "Employee Training"
- MP 1.22, "Integrated Standards Based Safety Management Program"
- MP 3.9, "Publications Services"
- MP 4.1, "Environmental Assurance"
- MP 4.4, "Radiological Protection"
- MP 4.5, "Nuclear Safety"
- MP 4.7, "Occupational Safety Policy"

- MP 4.12, "Emergency Preparedness"
- MP 4.13, "Reporting and Investigating of Environmental, Health, Safety, Security, Quality, or Programmatic Events, Conditions, and Concerns"
- MP 4.15, "Industrial Hygiene"
- MP 4.20, "Conduct of Operations"
- MP 5.5, "Site and Facilities Management"

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2. Spent Fuel Storage Division Administrative Manual (U). WSRC Procedure Manual RD-1, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.

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RDP 6.01, "Reactor Personnel Selection, Qualification, and Training (U)," Rev. 5, August 1994.

RDP 6.07, "Non-Reactor Nuclear Facility Personnel Selection, Qualification, and Training (U)," Rev. 0, June 1994.
3. Quality Assurance Manual (U). WSRC Procedure Manual 1Q, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.

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4. Generic Safety Analysis Report - Quality Assurance, Chapter 14. G-SAR-G-0014, Rev. 0, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC, March 1995.
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MP 1.2, "Management Policies, Requirements, and Procedures System," Rev. 2, November 1996.

MP 1.10, "Employee Communications," Rev. 2, January 1996.

MP 1.11, "Open Communication of Employee Concerns," Rev. 3, September 1996.

MP 1.18, "Employee Training," Rev. 4, January 1996.

MP 3.9, "Publications Services," Rev. 3, June 1996.

MP 4.1, "Environmental Assurance," Rev. 4, July 1996.

MP 4.4, "Radiological Protection," Rev. 4, August 1, 1995.

MP 4.5, "Nuclear Safety," Rev. 6, September 1996.

MP 4.7, "Occupational Safety Policy," Rev. 4, April 1997.

MP 4.12, "Emergency Preparedness," Rev. 0, April 1989.

MP 4.13, "Reporting and Investigating of Environmental, Health, Safety, Security, Quality, or Programmatic Events, Conditions, and Concerns," Rev. 4, August 1996.

MP 4.15, "Industrial Hygiene," Rev. 2, August 1995.

MP 4.19, "Lessons Learned Program," Rev. 2, August 1996.

MP 4.20, "Conduct of Operations," Rev. 2, April 1996.

MP 5.5, "Site and Facilities Management," Rev. 2, August 1994.

9. Management Requirements and Procedures (U). WSRC Procedure Manual 1B, Westinghouse Savannah River Company, Savannah River Site, Aiken, SC.

MRP 4.14, "Lessons Learned Program," Rev. 0, August 1993.

17.6 FIGURES

Figure 17.3-1 RBOF/RRF Organization

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