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Abstract The Cold Vacuum Drying (CVD) Facility Annex B Final Safety Analysis Report is part of the overall Spent Nuclear Fuel Project FSAR which is a multi volume document. This volume contains information specific to the CVD Facility. All topics have been addressed in this volume in accordance with DOE STD 3009 94 Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports.

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Volume 3

ANNEX B

COLD VACUUM DRYING FACILITY
FINAL SAFETY ANALYSIS REPORT

HNF-3553
Revision 0

November 1999
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EXECUTIVE SUMMARY

BE 1 FACILITY BACKGROUND AND MISSION

The U.S. Department of Energy (DOE) established the Spent Nuclear Fuel (SNF) Project to address safety and environmental concerns associated with deteriorating SNF presently stored under water in the Hanford Site K Basins, which are located in the 100 K Area near the Columbia River. Recommendations for a series of projects to construct and operate systems and facilities to manage the safe removal of K Basins fuel were made in WHC-EP-0830, *Hanford Spent Nuclear Fuel Recommended Path Forward,* and its subsequent update, WHC-SD-SNF-SP-005, *Hanford Spent Nuclear Fuel Project Integrated Process Strategy for K Basins Fuel.* The integrated process strategy recommends the following steps:

- Fuel preparation activities at the K Basins, including removing the fuel elements from their K Basins canisters, separating fuel particulate from fuel elements and fuel fragments greater than 0.25 in. in any dimension, removing excess sludge from the fuel fragments by means of flushing, as necessary, and packaging the fuel into multi-canister overpacks (MCOs);

- Transportation of MCOs loaded with SNF from the K Basins to the Cold Vacuum Drying Facility (CVDF), a new facility in the 100 K Area.

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Removal of free water from the fuel performed at the CVDF by draining of bulk water from the MCO and subsequent vacuum drying to remove remaining bulk water and sufficient portions of the hydration water, evacuation and backfilling of the MCO with an inert gas (helium), followed by sealing and leak testing of the MCO.

Dry shipment of fuel from the CVDF to the Canister Storage Building (CSB), a new facility in the 200 East Area.

Interim storage of the MCOs in the CSB until a suitable long-term repository is established.

The SNF Project Final Safety Analysis Report (FSAR) is a multi-volume document. Volume 1 contains SNF Project information applicable to all project facilities. Volume 2 contains information specific to the CSB. This volume, Volume 3, contains information specific to the CVDF. This volume represents the CVDF design as of June 7, 1999.

The purpose of Volume 3 is to provide the basis for authorization to operate the CVDF. The scope of this report includes the operating and support structures and equipment required for receipt cold vacuum drying, and preparation for dry shipment of the MCOs to the CSB. All topics identified in DOE-STD-3009-94, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports, are addressed, including descriptions of site and facility design, hazard and accident analyses, safety-class and safety-significant equipment, derivation of technical safety requirements, prevention of inadvertent criticality, and other areas of facility design and operational programs. This report also follows the guidance presented and approved in HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward.

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BE 2 FACILITY OVERVIEW

The CVDF is a new facility located to the west of the K Basins in the 100 K Area of the Hanford Site. The CVDF includes four adjoining areas: the process bay area, the process support area, the process water tank room, and the administrative building. The process bay area (60 ft by 150 ft) contains four operational process bays and one unused bay. The process bay area has been constructed using a steel frame with attached concrete panels, and each functional process bay contains a second-level mezzanine. Immediately adjacent and contiguous to the process bay area is the process support area (20 ft by 150 ft), a steel-framed, two-story metal building that encloses the traffic corridor, process bay support rooms, and the second floor mechanical equipment room. Immediately adjacent to the process bay area on the north side is a single-story concrete and structural steel building that encloses the process water tank room (20 ft by 60 ft), and a single-story, preengineered metal building that encloses office and change room functions.

Each operational process bay contains a process equipment skid, a safety-class helium system, a process hood, and a process bay recirculation heating, ventilation and air conditioning (HVAC) system. Each process equipment skid contains a vacuum and purge system and a tempered water (annulus) system.

The CVDF interfaces with the 100 K Area, Hanford Site infrastructure services, and the CSB. The CVDF operation interfaces with K Basins operations by receiving cask–MCO packages for processing. Water removed from the MCO and water used for system flushes is cleaned and transported by tanker truck to the K West Basin for further processing.

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Service water (filtered river water) is supplied to the CVDF fire protection system from the 100 K Area service water system. Potable water is supplied to the CVDF from the 100 K Area sanitary water loop. A feed line from Site utilities supplies the CVDF electrical systems. Security fire response, and medical services are provided as sitewide services. The CVDF also interfaces with the CSB operation when the cask-MCO packages are shipped to the CSB after the cold vacuum drying process has been completed.

BE 3 FACILITY HAZARD CLASSIFICATION

A final hazard categorization of the CVDF facility was determined from the results of the final accident analyses performed for the facility. These analyses provided an estimate of the material at risk for the hazard categorization performed in accordance with DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480 23 Nuclear Safety Analysis Reports*, Attachment 1 and with the guidance of HNF-PRO-704, *Hazard and Accident Analysis Process*, Appendix B.

Consistent with DOE-STD-1027-92, the CVDF is classified as a hazard category 2 nuclear facility and is based on the material at risk chosen to bound the material at risk from the worst-case unmitigated thermal runaway accident. These CVDF material quantities were compared against the threshold quantities contained in DOE-STD-1027-92.

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BE 4 SAFETY ANALYSIS OVERVIEW

The scope of the hazard analysis included all normal, intended CVDF operations for MCO receipt, cold vacuum drying, and preparation for dry shipment to the CSB. The analysis included reviews of draft operations, flow diagrams, and the facility final design. CVDF operations considered were those involved with bringing the transportation cask containing an MCO into the facility on the transporter, removing the cask lid, hooking up the processing equipment to the MCO, conducting the cold vacuum drying processing activities, conducting a series of final dryness tests, inerting the MCO with helium, disconnecting the processing equipment, replacing the cask lid, and transporting the MCO from the facility.

The hazard analysis involved the identification of hazard sources, hazardous conditions, potential accident scenarios and their initiators, and preliminary assessments of event frequencies and consequences. Hazards were identified by form and location and represent a complete spectrum of events that could occur throughout the facility.

The high-risk events that pose a challenge to offsite and onsite radiological dose evaluation guidelines were selected from the hazard analysis for further quantitative evaluation as design basis accidents (DBAs). These accidents are presented in detail in Chapter B3.4. Each DBA analyzed represents a bounding case for a category of hazards and accidents. The following six major DBA events are evaluated:

- Gaseous release
- Liquid release
- MCO external hydrogen explosion
- MCO internal hydrogen explosion
- MCO thermal runaway reaction
- MCO overpressurization
Four receptors were evaluated in the DBA analysis

- Hanford Site boundary receptor (offsite) defined at a distance of 10,090 m (33,100 ft) from the CVDF
- Collocated worker receptor (onsite), defined at a distance of 100 m (328 ft) from the CVDF
- Near bank of the Columbia River receptor (onsite), defined at a distance of approximately 650 m (2,130 ft) from the CVDF
- 100 Area Fire Station receptor (onsite) defined at a distance of approximately 3,750 m (12,300 ft) from the CVDF

Consequences to the Hanford Site boundary receptor and collocated worker receptor were compared against defined release limits and risk evaluation guidelines and were used for the selection of safety-class and safety-significant features, respectively. Consequences for receptors at the near bank of the Columbia River and at the 100 Area Fire Station also were calculated for the purpose of identifying any additional measures considered necessary to reduce the dose to individuals at these receptor locations. No additional safety features have been identified based on the consequences calculated for these locations. Safety-class preventive features and safety-significant preventive and mitigative features identified for the DBAs serve to reduce the dose consequences at all of the receptor locations.

BE 5 ORGANIZATIONS

Fluor Daniel Hanford, Incorporated is responsible to the DOE for the planning, integrating, and managing of SNF Project activities, including programs, projects, and operations.
Fluor Daniel Hanford is supported by subcontractors, including Duke Engineering and Services Hanford, Incorporated. Organizational responsibilities related to the SNF Project are summarized in the executive summary of Volume 1 and described in detail in Chapter 170 of that volume.

BE 6 SAFETY ANALYSIS CONCLUSIONS

The safety analysis conclusions for the CVDF DBAs are presented in detail in Chapter B30, Section B3 4 2 of this volume and are summarized as follows.

**Gaseous release** The bounding unmitigated scenario for this accident category describes a pressurized release of helium gas and entrained contaminated particulate through a process line leak. The unmitigated consequences of this event do not exceed the offsite release limits but do exceed the onsite risk evaluation guidelines. No safety-class features are required to mitigate this event. Safety-significant features selected to mitigate this event include portions of the process general supply/exhaust HVAC system and process bay local exhaust HVAC and process vent system and differential pressure alarms for the process bays and process water tank room. Mitigated consequences of this event are well below both offsite release limits and onsite risk evaluation guidelines. For a detailed description of safety features selected to prevent or mitigate other events within this accident category, see Section B3 4 2 1.

**Liquid release** The bounding unmitigated scenario for this accident category describes a pressurized leak of water and entrained contaminated particulate from the process water conditioning piping. The unmitigated consequences of this event do not exceed the offsite release limits but do exceed the onsite risk evaluation guidelines. No safety-class features are required to mitigate this event. Safety-significant features selected to mitigate this event include portions of the process general supply/exhaust HVAC system (ductwork and high-efficiency particulate air [HEPA] filters for the process water tank room) and the process water tank room differential pressure alarm. Mitigated consequences of this event are well below both offsite release limits.
and onsite risk evaluation guidelines. For a detailed description of safety features selected to prevent or mitigate other events within this accident category, see Section B3.4.2.2

**MCO external hydrogen explosion** The bounding unmitigated scenario for this accident category describes accumulation of hydrogen outside an MCO when it is vented from the MCO into the local exhaust process ventilation system and mixed with air, followed by ignition and explosion of the hydrogen gas. The unmitigated consequences of this event do not exceed the offsite release limits but do exceed the onsite risk evaluation guidelines. No safety-class features are required to prevent or mitigate this event. Safety-significant features selected to prevent this event include portions of the process bay local exhaust HVAC and process vent system (ductwork and HEPA filters) and special tools to limit the cask vent flow rate. Because the selected safety features prevent and mitigate this event, both offsite release limits and onsite evaluation guidelines are satisfied. For a detailed description of safety features selected to prevent or mitigate other events within this accident category, see Section B3.4.2.3

**MCO internal hydrogen explosion** The bounding unmitigated scenario for this accident category describes the ignition and explosion of a hydrogen–air mixture inside an MCO. The unmitigated consequences of this event do not exceed the offsite release limits but do exceed the onsite risk evaluation guidelines. No safety-class features are required to prevent or mitigate this event. However, some safety-class features preventing the MCO thermal runaway reaction and overpressurization events (i.e., multiple safety functions to detect process upsets, the safety-class helium system, portions of the tempered water [annulus] system) also prevent this accident, but in a safety-significant role. Because the designated safety features prevent and mitigate this event, both offsite release limits and onsite risk evaluation guidelines are satisfied. For a detailed description of safety features selected to prevent or mitigate other events within this accident category, see Section B3.4.2.4

**MCO thermal runaway reaction** The bounding scenario for this accident category describes an accident that is initiated by a reduction of heat removal from an MCO.
This condition results in the uncontrolled escalation of the chemical reaction within the MCO, resulting in high temperatures there. If unmitigated, the high temperatures of this scenario could lead to a continuous release of gas and contaminated particulate for an extended period of time. The unmitigated consequences of this event challenge the offsite release limits and exceed the onsite risk evaluation guidelines. Safety-class features selected to prevent this event include safety features to detect process upsets, the safety-class helium system purge and isolation, and portions of the tempered water (annulus) system. Because these safety-class features prevent this event, both offsite release limits and onsite risk evaluation guidelines are satisfied. For a detailed description of safety features selected to prevent or mitigate other events within this accident category, see Section B3.4.2.5

**MCO overpressurization** The bounding unmitigated scenario for this accident category describes overpressurization of an isolated MCO with no pressure relief. The pressure in an isolated MCO increases with the formation of hydrogen gas as a product of the uranium-water reaction. The MCO internal pressure continues to increase until the MCO pressure boundary is breached or until the fuel or water are completely consumed. The overpressurization leads to a pressurized release of gas and contaminated particulate followed by an extended period of slow continuous release driven by the continued oxidation of the uranium inside the MCO. The unmitigated consequences of this event exceed both the offsite release limits and onsite risk evaluation guidelines. Safety-class features selected to prevent this event include multiple safety features to detect process upsets, the safety-class helium system, the 30 lb/in² gauge vent line, the 150 lb/in² gauge rupture disk, and portions of the tempered water (annulus) system. These safety-class features reduce the frequency and mitigate the occurrence of this event to well within the offsite release limits. Additional safety significant features for confinement and filtration are identified to mitigate the onsite consequences to well below the onsite risk evaluation guidelines. For a detailed description of other safety features selected to prevent or mitigate other events within this accident category, see Section B3.4.2.6.
The following safety-class and safety-significant features are relied upon in the facility safety bases:

- Vacuum purge system
  - Isolation valves with instrument air filters (safety class)
  - Deionized water isolation valves with instrument air filters (safety class)
  - Isolation piping and flex hoses (safety class)
  - MCO connectors (safety class)
  - Pressure instruments (safety class)
  - 30 lb/in² vent path (safety class)

- General-service helium system
  - Safety relief valves (two) (safety class)
  - MCO helium flow transmitter trip (safety significant)
  - Isolation valve with instrument air filters (safety class)
  - Isolation piping (safety class)

- Process water conditioning system
  - Ion exchange modules (safety class)
  - Isolation valve with instrument air filters (safety class)
  - Isolation piping and flex hose (safety class)
  - Process water transfer line in process bays (safety significant)

- Process bay local exhaust HVAC and process vent system
  - Exhaust fans and plenum (safety significant)
Annex B — Cold Vacuum Drying Facility

- HEPA filter (safety significant)
- Ductwork (safety significant)
- Low flow alarm switch (safety significant)
- Hood isolation damper (fail closed) (safety significant)
- Cask venting orifice (safety significant)
- Cask vent jumper tool (safety significant)
- Cask venting valve (safety significant)
- Cask venting flow interlock (safety significant)
- MCO vent jumper tool (safety significant)

- Process general supply/exhaust HVAC system
  - HEPA filter (safety significant)
  - Exhaust ductwork (safety significant)
  - Isolation dampers (process bays) (safety significant)

- Process bay recirculation HVAC system
  - Isolation dampers (outside air inlets) (safety significant)

- Reference air system
  - Reference air header (safety significant)
  - Differential pressure alarms (safety significant)

- Instrument air system
  - Instrument air reservoirs (safety significant)
  - Air reservoir pressure gauges (safety significant)
Standby electrical power
  
  - Diesel generator (safety significant)
  - Local exhaust restart circuit (safety significant)

BE 7 SAFETY ANALYSIS REPORT ORGANIZATION

The CVDF FSAR is based on the format and content guidance of DOE-STD-3009-94\(^7\) and the requirements of DOE Order 5480 23, *Nuclear Safety Analysis Reports*\(^7\). This report also includes content guidance from NRC Regulatory Guide 3 26, *Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants*,\(^8\) as a result of the DOE regulatory policy described in HNF-SD-SNF-DB-003.\(^4\)

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<td>AC</td>
<td>Administrative Control</td>
</tr>
<tr>
<td>ALARA</td>
<td>as low as reasonably achievable</td>
</tr>
<tr>
<td>ARF</td>
<td>airborne release fraction</td>
</tr>
<tr>
<td>ARM</td>
<td>area radiation monitor</td>
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<tr>
<td>ARR</td>
<td>airborne release rate</td>
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<tr>
<td>ASHRAE</td>
<td>American Society of Heating Refrigerating and Air-Conditioning Engineers</td>
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<tr>
<td>ATC</td>
<td>automatic temperature control</td>
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<tr>
<td>BDBA</td>
<td>beyond design basis accident</td>
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<td>BED</td>
<td>building emergency director</td>
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<td>BEP</td>
<td>building emergency plan</td>
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<td>CAM</td>
<td>continuous air monitor</td>
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<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
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<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
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<tr>
<td>D&amp;D</td>
<td>decontamination and decommissioning</td>
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<tr>
<td>DBA</td>
<td>design basis accident</td>
</tr>
<tr>
<td>DBE</td>
<td>design basis earthquake</td>
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<td>DOE</td>
<td>U S Department of Energy</td>
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<tr>
<td>DOE-RL</td>
<td>U S Department of Energy Richland Operations Office</td>
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<tr>
<td>DPIT</td>
<td>differential pressure-indicating transmitter</td>
</tr>
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<td>EPZ</td>
<td>emergency planning zone</td>
</tr>
<tr>
<td>ERO</td>
<td>Emergency Response Organization</td>
</tr>
<tr>
<td>ERPG</td>
<td>Emergency Response Planning Guideline</td>
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<td>FHA</td>
<td>fire hazard analysis</td>
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<td>FSAR</td>
<td>final safety analysis report</td>
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<tr>
<td>HCI</td>
<td>human–computer interface</td>
</tr>
<tr>
<td>HEPA</td>
<td>high-efficiency particulate air (filter)</td>
</tr>
<tr>
<td>HFE</td>
<td>human factors engineering</td>
</tr>
<tr>
<td>HGP</td>
<td>Hanford Generating Project</td>
</tr>
<tr>
<td>HMI</td>
<td>human–machine interface</td>
</tr>
<tr>
<td>HSI</td>
<td>human–system interface</td>
</tr>
<tr>
<td>HVAC</td>
<td>heating, ventilation and air conditioning</td>
</tr>
<tr>
<td>IC</td>
<td>incident commander</td>
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<tr>
<td>ICP</td>
<td>Incident Command Post</td>
</tr>
<tr>
<td>IXM</td>
<td>ion exchange module</td>
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<td>KE</td>
<td>K East</td>
</tr>
<tr>
<td>KW</td>
<td>K West</td>
</tr>
<tr>
<td>LCO</td>
<td>Limiting Condition for Operation</td>
</tr>
<tr>
<td>LCS</td>
<td>Limiting Control Setting</td>
</tr>
<tr>
<td>MAR</td>
<td>material at risk</td>
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<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
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<td>MCS</td>
<td>monitoring and control system</td>
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<td>Multiple Launch Rocket System</td>
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<tr>
<td>mR</td>
<td>milli-roentgen</td>
</tr>
<tr>
<td>MSLD</td>
<td>mass spectrometer leak detector</td>
</tr>
<tr>
<td>NRC</td>
<td>U.S. Nuclear Regulatory Commission</td>
</tr>
<tr>
<td>ONC</td>
<td>Occurrence Notification Center</td>
</tr>
<tr>
<td>PAG</td>
<td>protective action guide</td>
</tr>
<tr>
<td>PES</td>
<td>process equipment skid</td>
</tr>
<tr>
<td>PLC</td>
<td>programmable logic controller</td>
</tr>
<tr>
<td>POC</td>
<td>Patrol Operations Center</td>
</tr>
<tr>
<td>PUREX</td>
<td>Plutonium-Uranium Extraction (Facility)</td>
</tr>
<tr>
<td>PWC</td>
<td>process water conditioning</td>
</tr>
<tr>
<td>QARD</td>
<td>quality assurance requirements and description</td>
</tr>
<tr>
<td>RF</td>
<td>respirable fraction</td>
</tr>
<tr>
<td>RFP</td>
<td>request for proposal</td>
</tr>
<tr>
<td>RGM</td>
<td>residual gas monitoring</td>
</tr>
<tr>
<td>SCHe</td>
<td>safety-class helium</td>
</tr>
<tr>
<td>SCIC</td>
<td>safety-class instrumentation and control</td>
</tr>
<tr>
<td>SMP</td>
<td>contaminated water sampling and analysis</td>
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<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
</tr>
<tr>
<td>SPR</td>
<td>single pass reactor</td>
</tr>
<tr>
<td>SSC</td>
<td>structure system, and component</td>
</tr>
<tr>
<td>TSR</td>
<td>technical safety requirement</td>
</tr>
<tr>
<td>UBC</td>
<td>Uniform Building Code</td>
</tr>
<tr>
<td>UPS</td>
<td>uninterruptible power supply</td>
</tr>
<tr>
<td>VPS</td>
<td>vacuum purge system</td>
</tr>
<tr>
<td>WNP</td>
<td>Washington Nuclear Plant</td>
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</table>

8-4-4 8-hour initial vacuum cycle, 4-hour subsequent vacuum cycles 4-hour return to pressure between vacuum cycles

ISO & PURGE  the MCO is isolated by closure of the VPS general-service helium system, PWC system, and deionized water isolation valves, and the SCHe system is actuated
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CHAPTER B1 0

SITE CHARACTERISTICS
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<td>Cold Vacuum Drying Facility</td>
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<td>DOE</td>
<td>U S Department of Energy</td>
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<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>HGP</td>
<td>Hanford Generating Project</td>
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<tr>
<td>KE</td>
<td>K East</td>
</tr>
<tr>
<td>KW</td>
<td>K West</td>
</tr>
<tr>
<td>MCO</td>
<td>multi-camster overpack</td>
</tr>
<tr>
<td>MLRS</td>
<td>Multiple Launch Rocket System</td>
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<tr>
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<td>U S Nuclear Regulatory Commission</td>
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<td>structure, system, and component</td>
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<td>Washington Nuclear Plant</td>
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B1 0 SITE CHARACTERISTICS

B1 1 INTRODUCTION

The Cold Vacuum Drying Facility (CVDF) is located in the 100 K Area of the U.S. Department of Energy (DOE) Hanford Site. The CVDF will be used to remove free water from spent nuclear fuel (SNF) taken from the K West (KW) and K East (KE) fuel storage basins and to vacuum dry the SNF before it is transferred to the Canister Storage Building in the 200 East Area. This chapter describes the characteristics of the 100 K Area, including the site on which the CVDF is located, and supports the hazard analysis and accident analyses presented in Chapter B3. A detailed description of the CVDF structures and systems is provided in Chapter B2. This chapter and Chapter 10 of the SNF Project Final Safety Analysis Report (FSAR) contain information related to regional and Hanford Site characteristics.

B1 2 REQUIREMENTS

The requirements that establish the basis for siting the CVDF are identified in Section 1.2 of the SNF Project FSAR. Some of the requirements listed in Section 1.2 of the SNF Project FSAR were not in existence at the time the CVDF was designed. The impact of these later orders and standards is reflected in WHC-SD-SNF-DB-010, Cold Vacuum Drying System Natural Phenomena Hazards. A discussion of U.S. Nuclear Regulatory Commission (NRC) equivalency requirements is provided in Section 1.2 of the SNF Project FSAR.

In addition to the requirements identified in Section 1.2 of the SNF Project FSAR, the following industry standards are applicable to the CVDF safety basis:

- ASCE 7-95, 1995, Minimum Design Loads for Building and Other Structures

B1 3 SITE DESCRIPTION

The following sections address the geography, demography, and regional land and water use of the area encompassed by, and surrounding, the 100 K Area. The information contained in these sections was obtained primarily from DOE/EIS-0245F, Management of Spent Nuclear Fuel from the K Basins at the Hanford Site, Richland, Washington.
B1 3 1 Geography

**B1 3 1 1 Hanford Site Vegetation**  Section 1 3 1 1 of the SNF Project FSAR provides information applicable to all SNF Project facilities

**B1 3 1 2 Hanford Site Facilities**  Section 1 3 1 2 of the SNF Project FSAR provides information applicable to all SNF Project facilities

**B1 3 1 3 Boundaries for Evaluation of Accident and Effluent Release Limits**  As stated in Section 1 3 1 3 of the SNF Project FSAR, activities at the 100 K Area are within the DOE-controlled zone. DOE has directed that consequences of accident releases to collocated workers and to the public (offsite receptor) be calculated at the following four locations (Sellers 1996) (see Section B3 4 1 2)

- Receptors located 100 m from the CVDF
- Receptors located at the near bank of the Columbia River (about 650 m, the nearest point of uncontrolled public access)
- Receptors located at the fire station (about 3,750 m to the east-southeast)
- Receptors located at the Hanford Site boundary (about 10,000 m to the west of the K West Basin)

Consequences of accident releases from the CVDF to collocated workers are calculated in Section B3 4 2 at 100 m from the point of release in accordance with approved procedures. Routine and accidental releases to the public (offsite receptor) are calculated at the Hanford Site boundary (North Wahluke Slope) shown in Figure B1-1. This Hanford Site boundary is also the location of the controlled area boundary, as the term is defined in Title 10 Code of Federal Regulations Part 72 'Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste' (10 CFR 72), Section 72 106 "Controlled Area of an ISFSI or MRS."

Doses at the near bank of the Columbia River were calculated for the purpose of identifying any additional measures considered necessary to reduce the dose to individuals at this receptor location as stated in Section B3 4 1 2. For Highway 240 to the south and west, DOE and its contractors can control access during emergency and accident conditions. This access control meets the requirements of 10 CFR 72 106(b). Before changes to the Site boundaries are placed into effect by the proper authorities, the calculations in this FSAR will be reviewed (and reanalyzed if required) in accordance with DOE Order 5480 21 "Unreviewed Safety Questions"
B1 3 2 Demography

Section 1 3 2 of the SNF Project FSAR provides a description of the onsite and offsite demography.

B1 4 ENVIRONMENTAL DESCRIPTION

B1 4 1 Meteorology

Section 1 4 1 of the SNF Project FSAR provides meteorology information applicable to all SNF Project facilities.

The specific air transport factors used for accident consequence analyses, and the basis for their calculation and use, are discussed in detail in HNF-SD-SNF-TI-059, A Discussion on the Methodology for Calculating Radiological and Toxicological Consequences for the Spent Nuclear Fuel Project at the Hanford Site, and in Section B3 4 1 2. To support the accident analyses of Section B3 4 2, air transport factors (X/Q and X/Q'), representing the dilution of a contaminant by atmospheric turbulence and diffusion as the contaminant travels downwind, were calculated (HNF-SD-SNF-TI-059). The symbol X/Q is the ratio of the average air concentration at the receptor to the average release rate at the release point. It is used to assess potential radiological dose and noncorrosive chemical concentration at downwind locations. The symbol X/Q' is the normalized peak air concentration at the center of a puff divided by the quantity released, and is used to assess the consequences to a receptor for corrosive chemicals. The X/Q's for the analyses were calculated using joint frequency distribution data so as to be exceeded only 0.5% of the time (99.5% percentile) for each sector, or to be exceeded only 5% of the time (95% percentile) for data from all sectors combined (the greater of the two calculated values is used in the analyses).

The GXQ computer code, Version 4.0 (WHC-SD-GN-SWD-30002), was used to generate X/Q and X/Q' values. GXQ incorporates the methods described in NRC Regulatory Guide 1 145 Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants. The specific air transport factors used for accident consequence analyses, and the basis for their calculation and use, are discussed in detail in HNF-SD-SNF-TI-059 and in Section B3 4 1 2. Also see supporting information in Table B3-7 for the atmospheric transport factors used in accident analyses for the CVDF. Supporting information is also provided in Section 1 4 1 2 8 of the SNF Project FSAR.

The wind rose data (Figure B1-2 and Figure B1-3) indicate that winds from the west sector occur most frequently (nearly 16% of the time). That is, the emissions are transported toward the east sector. Winds out of the west-northwest and west-southwest also occur with a relatively high frequency (12% and 10%, respectively). Information provided in these figures corresponds to the data presented in HNF-SD-SNF-TI-059.
B1 4 2 Hydrology

Section 1 4 2 of the SNF Project FSAR provides Hanford Site hydrology information that is applicable to all SNF Project facilities.

B1 4 2 1 Surface Water Section 1 4 2 1 of the SNF Project FSAR provides surface water information.

B1 4 2 2 Vadose Zone Section 1 4 2 2 of the SNF Project FSAR provides a definition of the vadose zone.

B1 4 2 3 Aquifers The 100 Area aquifer systems are discussed in Section 1 4 2 3 of the SNF Project FSAR.

B1 4 3 Geology

A description of the Hanford Site geology which applies to all SNF Project facilities, is provided in Section 1 4 3 of the SNF Project FSAR.

B1 4 3 1 Physiographic Setting of the Hanford Site See Section 1 4 3 1 of the SNF Project FSAR for information on the physiographic characteristics applicable to the CVDF.

B1 4 3 2 Stratigraphy See Section 1 4 3 2 of the SNF Project FSAR for information on the stratigraphy of the Pacific Northwest and the Hanford Site that is applicable to the CVDF.

B1 4 3 3 History of Cataclysmic Flooding in the Pasco Basin See Section 1 4 3 3 of the SNF Project FSAR for information on cataclysmic floods that is applicable to the CVDF.

B1 4 3 4 Geologic Structures of the Columbia Basin and Hanford Site See Section 1 4 3 4 of the SNF Project FSAR for information on geologic structures that is applicable to the CVDF.

B1 4 3 5 Geology of the 100 K Area The 100 K Area and vicinity is underlain by the Columbia River Basalt Group and intercalated Ellensburg Formation, the Ringold Formation, the Hanford formation and Holocene deposits. The 100 K Area is near the axis of the Wahluke syncline. The Columbia River Basalt Group lies approximately 527 ft beneath the 100 K Area. Overlying the Columbia River Basalt Group is the Ringold Formation (Figure B1-4).

The Ringold Formation is about 493 ft thick and consists of the fluvial gravels and sands of units A, B, C and E lacustrine and fluvial overbank deposits, and paleosols (Figures B1-4, B1-5, and B1-6). Unit A, the lowermost unit, is approximately 7 m (23 ft) thick and consists of fluvial gravel facies grading upward into sand associations. Overlying unit A is the Lower Mud unit. The Lower Mud unit is approximately 105 ft thick. Unit B is approximately 92 ft thick and consists predominantly of sand. Overlying unit B is a 209-ft-thick sequence of muds and sandy
muds typically displaying characteristics of paleosols and fluvial overbank deposits. The sequence has three parts: an upper and a lower part that are predominantly silt to sandy silt and a middle section of gravelly sand. The uppermost unit of the Ringold Formation at the 100 K Area is the coarse-grained unit E, which is predominantly composed of the fluvial gravel and fluvial sand facies. The unconfined portion of the uppermost aquifer system occurs in this unit at the 100 K Area. The known thickness of unit E ranges from 64 to 141 ft.

The Hanford formation at the 100 K Area is a wedge that decreases in thickness toward the Columbia River. It is approximately 120 to 130 ft thick. The Hanford formation pinches out from southeast to northwest across the 100 K Area. The gravel-dominated facies predominates in the Hanford formation throughout the 100 K Area. Boulder gravel is often found in the upper 20 to 50 ft. The sand-dominated facies occurs locally in a few intervals, but it is not thick enough or extensive enough to correlate from borehole to borehole. The silt-dominated facies has not been identified in the 100 K Area.

Holocene deposits in the study area are dominated by Columbia River deposits and eolian deposits. Columbia River deposits consist of gravels and coarse-grained sands deposited in channels and silts and fine sands deposited in the overbank area. Eolian deposits consist dominantly of less than 3 ft of silty, fine-grained sands that blanket much of the area except in locations where they were removed for construction purposes. In many locations eolian deposits are only a thin blanket (less than 1 ft thick).

The 100 K Area is geologically different than surrounding areas (100 B, 100 C, and 100 N Areas) because the Ringold Formation is exposed, not only along the banks of the Columbia River, but also from the river to 1,200 ft or more away from the river to the southeast.

The Wahluke syncline is the principal structural unit that contains the 100 Areas and the CVDF. The Wahluke syncline is an asymmetric and relatively flat-bottomed structure similar to the Cold Creek syncline. The northern limb dips gently (approximately 5°) to the south. The steepest limb is adjacent to the Umtanum Ridge-Gable Mountain structure.

The 100 K Area lies along the axis of the Wahluke syncline. The bedrock basalt surface tilts gently to the south in the area north of the 100 K Area. The basalt is tilted steeply to the north along the north flank of the Gable Butte portion of the Umtanum-Gable Mountain anticline.

The 100 K Area is about 1.2 mi north of the Umtanum-Gable Mountain thrust fault. The Umtanum-Gable Mountain fault is a reverse to thrust fault that extends over 62 mi along the north side of the Umtanum-Gable Mountain anticline. The Central Fault on the Gable Mountain segment of the Umtanum-Gable Mountain anticline is a capable fault (NUREG-0892, Supplement No. 1) and is 8 mi east of the 100 K Area.

**B1 4 3 6 Tectonic Development of the Hanford Site** See Section 1 4 3 6 of the SNF Project FSAR for information on geologic structures and faults that relate to Hanford Site seismic hazard.
analysis that is applicable to the CVDF. Also see WHC-SD-W236A-TI-002 Probabilistic Seismic Hazard Analysis DOE Hanford Site Washington.

B1 4 3 7 Contemporary Stress and Strain See Section 1 4 3 7 of the SNF Project FSAR for information on earthquake activity, contemporary stress measurements and subsidence history that is applicable to the CVDF.

B1 4 3 8 Geologic Hazards

B1 4 3 8 1 Seismic Hazard Assessment The major source of earthquakes at the Hanford Site is swarm activity in the syncline of the Yakima Fold Belt. There are three general areas of significant swarm activity: the Wooded Island swarm area, the Coyote Rapids swarm area, and the Saddle Mountains swarm area.

The Coyote Rapids swarm area is located at the horn of the Columbia River between the 100 K and 100 N Areas. It occurs over no known geologic structure. The CVDF is located in the southern portion of this swarm. The swarm lies at the intersection of two paleoslopes that make a northeast-southwest trough extending from Spokane, Washington to the Columbia Gorge. This zone may be an old basement weakness zone, but there is no known reason for the swarm to occur in its present position. The largest earthquake recorded on the Hanford Site has a magnitude of 3.8 and was a part of the Coyote Rapids swarm area.

The mean seismic hazard curves for the 100 K Area are shown in Figures B1-7 and B1-8 and illustrate the contributions of individual folds to the hazard. The relative contribution of crustal and Cascadia Subduction sources at the 100 K Area is illustrated in Figure B1-9 and the relative contribution of the three crustal sources for the same location is shown in Figure B1-10.

The performance category 3 horizontal and vertical equal-hazard response spectra were developed for the CVDF site. These spectra are shown at 5% damping for performance category 3 in Figure B1-11. They are the enveloping spectra for the 200 Areas and the 100 K Area. More detail and additional damping values are presented in WHC-SD-W236A-TI-002. As stated in WHC-SD-SNF-DB-010, performance category 3 projects follow the requirements of DOE Order 5480.28 Natural Phenomena Hazards Mitigation as a minimum and supporting standard DOE-STD-1020-94, Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities, Chapter 2 and Appendix C. Equal-hazard 5% damped horizontal and vertical response spectra are anchored at 0.26 g horizontal. In-structure response spectra are generated for structures, systems, and components (SSCs) supported by the CVDF structure. This information is provided in Chapter B4.0 for the design of the CVDF SSCs.

B1 4 3 8 2 Volcanic Hazard Assessment Information about the volcanic hazards applicable to all SNF Project facilities is provided in Section 1 4 3 8 2 of the SNF Project FSAR.
B1 4 3 8 3 Subsurface Stability The CVDF is constructed on flood sediments, the youngest sediments being approximately 13,000 years old. There are no areas of potential surface or subsurface subsidence, uplift, or collapse except for the low geologic deformation. With the exception of the loose, surficial, wind-deposited silt, soils are competent and form good foundations. Geotechnical studies have been completed in and around the 100 K Area (WHC-SD-NR-ER-093, Redpath 1994). The water table is approximately 75 ft below ground surface in the relatively compact and moderately well cemented unit of the Ringgold Formation.

B1 5 NATURAL PHENOMENA THREATS

Table B1-1 (Table 1 of WHC-SD-SNF-DB-010) summarizes the natural phenomena design loads to be used for the design of safety-class SSCs for the CVDF. The industry codes and standards used for the design, fabrication, and procurement of safety-significant SSCs are shown in Table B4-3. The design codes used for general service equipment follow industry acceptable design codes and standards. These SSCs are designed to codes, standards, regulations, and orders, as listed in Section 1.2 of the SNF Project FSAR. Each of these phenomena and the bases for the analysis values have been discussed in Section 1.4 of the SNF Project FSAR.

B1 6 EXTERNAL HUMAN-GENERATED THREATS

This section identifies and investigates specific potential human-generated threats to CVDF operation. Threats to the CVDF from human activities that are not known at this time will be evaluated when identified by the Unreviewed Safety Question process.

B1 6 1 Aircraft Activity

The methodology used in the aircraft activity analysis is provided in Section 1.6.1 and Figure 1-35 of the SNF Project FSAR. There are twelve active airports within a 35-mi radius of the CVDF (Beary 1997). Ten of these are small airports that serve only general aviation aircraft and are within 24 mi of the CVDF. The Richland Airport, 27 mi southeast of the CVDF, primarily supports general aviation operations, but two commercial freight carriers per day land and take off from the airport’s runways. The nearest airport with significant commercial and military air activity is the Tri-Cities Airport, 34 mi southeast of the CVDF.

DOE-STD-3014-96, Accident Analysis for Aircraft Crash into Hazardous Facilities, provides the methodology to conservatively evaluate and assess the significance of aircraft crash risk on facility safety. The CVDF meets the standard’s guidelines for applicability because it contains enough radioactive material to be classified as a hazard category 2 nuclear facility according to the criteria established in DOE-STD-1027-92, Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23 Nuclear Safety Analysis Reports.
Table B1-1 Cold Vacuum Drying Facility Safety-Class Natural Phenomena Design Loads

<table>
<thead>
<tr>
<th>Hazard</th>
<th>Load</th>
<th>Application documents</th>
</tr>
</thead>
</table>
| Seismic           | Performance Category 3 (0.26 g) equal hazard response spectra | DOE Order 5480 28<sup>b</sup>  
DOE Standard 1020 94  
Table 2 and Figure 1 |
| Straight wind     | 80 mi/h fastest mile at 30 ft             | ASCE 7 95<sup>d</sup>  
DOE STD 1020 94 (including missiles) |
| Tornado           | Wind speeds                               | NRC Standard Review Plan  
3.3.2 Tornado Loading |
|                   | 200 mi/h total                            |                                                            |
|                   | 160 mi/h rotational                       |                                                            |
|                   | 40 mi/h translational                     |                                                            |
| Volcanic ash      | 24 lb/ft<sup>2</sup> ground ash load     | NRC Standard Review Plan  
3.8.4 Other Seismic Category Structures |
| Flooding          | Columbia River 460 ft above mean sea level | ANSI/ANS 2 8 1992<sup>c</sup>  
NRC Standard Review Plan  
2.4.2 Floods |
|                   | Site drainage basin 7.4 in for           |                                                            |
|                   | 6 hour probable maximum precipitation     |                                                            |
|                   | Site drainage 9.2 in for                  |                                                            |
|                   | 6 hour probable maximum precipitation     |                                                            |
| Lightning         | Lightning protection shall be provided for facility as required by the code | NFPA 780<sup>e</sup> |
| Snow              | 20 lb/ft<sup>2</sup> ground load         | ASCE 7 95<sup>d</sup> |

Note: The wind speed criteria are based on recent revisions of the Washington Nuclear Plant 2 tornado criteria by the NRC (1) Parrish J V 1995  

WHC SD W236A T1 002 1997  Probabilistic Seismic Hazard Analysis DOE Hanford Site Washington Rev IA prepared by Geomatix Consultants Incorporated for Westinghouse Hanford Company Richland Washington
<sup>b</sup> DOE Order 5480 28  Natural Phenomena Hazards Mitigation U.S. Department of Energy Washington D.C
<sup>c</sup> DOE STD 1020 94  Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities U.S. Department of Energy Washington D.C
<sup>d</sup> ASCE 7 95 1995  Minimum Design Loads for Building and Other Structures American Society of Civil Engineers New York New York
<sup>e</sup> NUREG 0800 1981  Standard Review Plan U.S. Nuclear Regulatory Commission Washington D.C
<sup>g</sup> NFPA 780 1995  Lightning Protection Systems National Fire Protection Association Quincy Massachusetts
The approach used in DOE-STD-3014-96 includes the methodology to evaluate the frequency of aircraft impact into the facility. If the frequency of aircraft impact, calculated according to the methodology given in the standard, exceeds $10^{-6}$/yr, the analysis proceeds to a consideration of whether those aircraft that have a high impact frequency could actually cause damage resulting in potential releases from the facility. Using that methodology, the frequency of aircraft impact into the CVDF has been assessed (Beary 1997).

The methodology provided in DOE-STD-3014-96 considers two contributors to the overall frequency of aircraft crashes: (1) the frequency of crashes resulting from nearby airport activity (i.e., takeoffs and landings) and (2) the frequency of crashes during overflights of the facility. Only runways within about 24 mi of the facility are included on the crash location probability tables given in Appendix B of DOE-STD-3014-96. Activities on runways at greater distances are not considered potential contributors to the frequency of a crash at the facility location for any category of aircraft. For calculating the frequency of crashes from aircraft flying over the site, DOE-STD-3014-96 provides DOE site-specific frequency data, in crashes per square mile per year centered at the site for various classes of aircraft.

As stated above, there are 10 active airports within a 24-mi radius of the CVDF (Beary 1997). All operations at these airports involve general aviation aircraft. The nearest airports serving commercial or military aircraft are the Richland Airport, 27 mi southeast of the CVDF and the Tri-Cities Airport, 34 mi southeast. Table B1-2 lists the 10 airports within 24 mi of the CVDF and their distances from and orientation to the facility.

For both the nearby airports and the overflight activities, DOE-STD-3014-96 bases the calculation of frequency of aircraft crash into a facility on a "four factor formula" that considers the following:

1. The number of aircraft operations at nearby airports and overflying the site ($N$).
2. The probability that an aircraft will crash ($P$).
3. The conditional probability that, given a crash, the aircraft crashes into a one-square-mile area where the facility is located ($f(x,y)$).
4. The site-specific effective area of the facility ($A_{eff}$).

The formula is applied individually to each category of aircraft for both the nearby airports and the overflight activities. The overall frequency of aircraft crashes into the facility is the sum of the frequencies calculated for all categories of aircraft and for all operational modes.
The distances and orientations of the runways for the 10 nearby airports were compared to tables in Appendix B of DOE-STD-3014-96 to identify which of the flight sources contribute to the frequency that an aircraft will crash in a one-square-mile area in the vicinity of the CVDF. All of the airports exclusively support general aviation. The screening revealed that only the landing operations on runways 18 and 36 at the Mattawa Air Strip and on runway 27 at Desert Aire contribute to the crash frequency for nearby airport operations for the CVDF.

The Mattawa Air Strip reports 1,250 operations per year, and Desert Aire has approximately 12 operations per week or 624 operations per year. It was assumed that the operations were equally distributed between runways and that there are an equal number of takeoffs and landings. Table 1 of DOE-STD-3014-96 gives a crash rate for landings of representative fixed wing general aviation aircraft of \( 2 \times 10^{-5} \) per landing.

The conditional probability \( f(x,y) \), that a general aviation aircraft attempting to land on one of these runways will crash into a one-square-mile area containing the CVDF was taken from Table 5 of DOE-STD-3014-96. The values are \( 8.7 \times 10^{-5} \) for Mattawa runway 18, \( 1.8 \times 10^{-4} \) for Mattawa runway 36, and \( 5.6 \times 10^{-5} \) for Desert Aire runway 27.

Multiplying the number of operations per year, \( N \), the crash rate per operation, \( P \), and the conditional probability per square mile, \( f(x,y) \), gives a factor \( NPf(x,y) \), representing the contribution for each airport for general aviation operations. The values for the three flight
sources can be summed at this point because all involve the same category of aircraft (general aviation) and the same flight phase (landing). The value of NPf(x,y) for operations for the Mattawa runways is

\[
NPf(x,y)_1 = 2 \times 10^{-5} \frac{\text{crashes}}{\text{landing}} \left( \frac{1.25}{4} \frac{\text{landings/yr}}{\text{runway}} \times (8.7 \times 10^{-5} + 1.8 \times 10^{-4}) \frac{\text{impacts}}{\text{mi}^2} \right)
\]

\[
= 1.67 \times 10^{-6} \frac{\text{impacts/yr}}{\text{mi}^2}
\]

The value of NPf(x,y) for operations at Desert Aire runway is

\[
NPf(x,y)_2 = 2 \times 10^{-5} \frac{\text{crashes}}{\text{landing}} \left( \frac{624}{4} \frac{\text{landings/yr}}{\text{runway}} \times 5.6 \times 10^{-5} \frac{\text{impacts}}{\text{mi}^2} \right)
\]

\[
= 1.75 \times 10^{-7} \frac{\text{impacts/yr}}{\text{mi}^2}
\]

This factor will ultimately be multiplied by the effective area of the facility to give the contribution of operations at that airport to the overall annual frequency of aircraft crashes into the facility.

To evaluate the frequency of crashes that impact the CVDF from overflight operations, Tables 14 and 15 of DOE-STD-3014-96 give generic values of NPf(x,y) in crashes per square mile per year centered at various DOE sites, including the Hanford Site, for five categories of aircraft. The values are based on the historic record for aircraft crashes in the continental United States.

The site-specific values of NPf(x,y) for overflight operations given for the Hanford Site by aircraft category are as follows:

- General aviation \(1 \times 10^{-4}\) crashes/yr/mi\(^2\)
- Air carrier \(1 \times 10^{-7}\) crashes/yr/mi\(^2\)
- Air taxi \(1 \times 10^{-6}\) crashes/yr/mi\(^2\)
- Large military \(1 \times 10^{-7}\) crashes/yr/mi\(^2\)
- Small military \(4 \times 10^{-8}\) crashes/yr/mi\(^2\)

For both the nearby airport and the overflight operations, the NPf(x,y) for each category of aircraft is multiplied by an effective area, \(A_{\text{eff}}\), representing the ground surface area within which an unobstructed aircraft, were it to crash within the area, would impact the facility, either by flying or skidding into the facility. The effective area depends not only on the length, width, and
height of the facility, but also on the aircraft's wingspan, the flight path angle, the heading angle relative to the heading of the facility, and the length of the skid. Therefore, an effective area is calculated for each category of aircraft.

Formulas for calculating the effective area of the facility are given in Appendix B of DOE-STD-3014-96. The effective area has two terms: the effective fly-in area \( A_f \), and the effective skid area \( A_s \)

\[
A_{\text{eff}} = A_f + A_s
\]

and

\[
A_f = (WS + R) H\cot \phi + \frac{2 LWWS}{R} + LW
\]

\[
A_s = (WS + R) S
\]

where

- \( WS \) = aircraft wingspan
- \( R \) = length of the diagonal of the facility \((L^2 + W^2)^{0.5}\)
- \( H \) = facility height (35 ft for the CVDF)
- \( \cot \phi \) = mean of the cotangent of the aircraft impact angle
- \( L \) = facility length (180 ft for the CVDF)
- \( W \) = facility width (110 ft for the CVDF)
- \( S \) = aircraft skid distance (mean value)

The CVDF dimensions used in the calculation include the helium tanks outside the facility but not the administration wing. The values of \( WS \), \( \cot \phi \), and \( S \) are given in Tables 16, 17, and 18 of DOE-STD-3014-96 for the categories of aircraft. For the military aviation categories, different values for the mean of \( \cot \phi \) and the mean skid distance are given for takeoff and landing. DOE-STD-3014-96 recommends using the takeoff values for the calculations involving overflight operations. Table B1-3 presents the calculated effective areas for the CVDF for all the classes of aircraft.

Multiplying the \( NPf(x, y) \) factors for both the nearby airport and the overflight operations by the effective impact areas gives the annual frequency of impact into the facility for each class of aircraft. These values are given in Table B1-4.
### Table B1-3 Results of Effective Area Calculations for the Cold Vacuum Drying Facility

<table>
<thead>
<tr>
<th></th>
<th>General aviation</th>
<th>Commercial aviation</th>
<th>Military aviation</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Air carrier</td>
<td>Air taxi</td>
</tr>
<tr>
<td>WS</td>
<td>50</td>
<td>98</td>
<td>59</td>
</tr>
<tr>
<td>R (ft)</td>
<td>210 95</td>
<td>210 95</td>
<td>210 95</td>
</tr>
<tr>
<td>H (ft)</td>
<td>35</td>
<td>35</td>
<td>35</td>
</tr>
<tr>
<td>cotϕ</td>
<td>8 2</td>
<td>10 2</td>
<td>10 2</td>
</tr>
<tr>
<td>W (ft)</td>
<td>110</td>
<td>110</td>
<td>110</td>
</tr>
<tr>
<td>L (ft)</td>
<td>180</td>
<td>180</td>
<td>180</td>
</tr>
<tr>
<td>S (ft)</td>
<td>60</td>
<td>1,440</td>
<td>1,440</td>
</tr>
<tr>
<td>A_r+A_s</td>
<td>0 00429</td>
<td>0 0211</td>
<td>0 0184</td>
</tr>
</tbody>
</table>

### Table B1-4 Crash Frequencies for the Cold Vacuum Drying Facility

<table>
<thead>
<tr>
<th></th>
<th>NPf(x,y) (impacts/yr/m²)</th>
<th>A_{eff} (m²)</th>
<th>Crash frequency (per year)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Nearby airport operations (general aviation only)</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Mattawa Airfield</td>
<td>1 67 E-06</td>
<td>4 29 E-03</td>
<td>7 16 E-09</td>
</tr>
<tr>
<td>Desert Aire</td>
<td>1 75 E-07</td>
<td>4 29 E-03</td>
<td>7 51 E-10</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td></td>
<td>7 9 E-09</td>
</tr>
<tr>
<td><strong>Overflight operations</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>General aviation</td>
<td>1 E-04</td>
<td>4 29 E-03</td>
<td>4 29 E-07</td>
</tr>
<tr>
<td>Commercial aviation, air carrier</td>
<td>1 E-07</td>
<td>2 11 E-02</td>
<td>2 11 E-09</td>
</tr>
<tr>
<td>Commercial aviation, air taxi</td>
<td>1 E-06</td>
<td>1 84 E-02</td>
<td>1 84 E-08</td>
</tr>
<tr>
<td>Military aviation, large aircraft</td>
<td>1 E-07</td>
<td>1 84 E-02</td>
<td>1 84 E-09</td>
</tr>
<tr>
<td>Military aviation, small, low performance</td>
<td>4 E-08</td>
<td>7 67 E-03</td>
<td>3 07 E-10</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td></td>
<td>4 5 E-07</td>
</tr>
<tr>
<td><strong>Total all operations</strong></td>
<td></td>
<td></td>
<td>4 6 E-07</td>
</tr>
</tbody>
</table>
The major contributor to the frequency of aircraft impact into the CVDF is general aviation aircraft during in-flight operations. The overall frequency of aircraft impact from all sources is \(4.6 \times 10^{-4} \text{ /yr}\). \(\text{DOE-STD-3014-96}\) specifies that if the total impact frequency, calculated according to the method given in the standard, is less than \(10^{-4} \text{ /yr}\), the safety risk is below the level of concern. Therefore, it is not necessary to further calculate the probability that the impact will lead to a release from the facility or the consequences of such a release.

For rotary wing aircraft, see Section 1.6.1 of the SNF Project FSAR.

Medical evacuation helicopters' closest approach to the CVDF will be a landing pad at the 100 Area fire station, which is about 2.1 mi from the CVDF.

**B1.6.2 Other Transportation Accidents**

A discussion of the guidance for evaluating transportation accidents is provided in Section 1.6.2 of the SNF Project FSAR.

Figure B1-12 shows a rail spur that passes by the CVDF. This spur has not been used recently and will not be used in support of SNF Project activities. The switch that would route trains on the spur is normally locked in the position that would route trains away from the CVDF (see K Basins FSAR HNF-SD-WM-SAR-062).

Accidents that might occur on State Highway 240, such as explosions or toxic chemical releases, are judged to present a negligible risk to the CVDF because of the distance between the facility and the highway. At its closest approach, the distance is about 6 mi. NRC Regulatory Guide 1.78, *Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release*, provides useful guidance on evaluating chemicals stored or situated at distances greater than 5 mi from the facility. It states that they need not be considered because if a release occurs at such a distance, atmospheric dispersion will dilute and disperse the incoming plume to such a degree that there should be sufficient time for the operators to take appropriate action. In addition, the probability of the plume remaining within a given sector for a long time is quite small.

NRC Regulatory Guide 1.78 also provides useful guidance for the evaluation of potential accidents involving hazardous chemicals that might be shipped past the CVDF on Route 4 North and the Hanford Site railroad. NRC Regulatory Guide 1.78 does not require control room habitability analysis for shipments less frequent than 10/yr for truck traffic, 30/yr for rail traffic, or 50/yr for barge traffic. Neither the truck nor rail guidelines are exceeded for shipments of a quantity that could present a risk to the CVDF. Section B3.3 addresses this hazard from the perspective of safety of CVDF workers and capability to maintain the facility in a safe state. Barge shipment on the Columbia River does not routinely occur above the Port of Benton barge facility as discussed in Section 1.3.1 of the SNF Project FSAR.
B1 7 NEARBY FACILITIES

Accidents in certain facilities in the 100 Areas have the potential to impact the CVDF and its operations, as discussed in Section B1 7 1 Conversely, certain nearby facilities can potentially be affected by accidents in the CVDF, as discussed in Section B1 7 2

Threats to the CVDF from nearby facilities that are not known at this time will be evaluated when identified This also applies to those activities that have been identified but that may change significantly in terms of a potential increased risk to the facility These will be evaluated when identified by the Unreviewed Safety Question process

B1 7 1 Potential Effects from Nearby Facilities

Potential hazards to the CVDF from onsite or offsite hazardous operations or facilities are examined under three general classifications

1. Nonreactor nuclear and nonnuclear industrial facilities within 5 mi of the CVDF
2. Nuclear reactors within a 50-mi radius of the CVDF
3. Military activities

If the CVDF habitability is challenged by an event at a nearby facility, the appropriate actions (i.e., notifications, take-cover, emergency shutdown, evacuation) will be defined in the CVDF emergency response plan and emergency response procedures See Section B15 4 6 and Section 15 4 6 of the SNF Project FSAR for additional information on protective actions

B1 7 1 1 Hazards to the Cold Vacuum Drying Facility from Nonreactor Nuclear Facilities

B1 7 1 1 1 K Basins The greatest risk for release of radionuclides in the 100 Areas exists with the K Basins The basins are in the 105-KE and 105-KW reactor buildings The storage basins were modified and reactivated to provide interim storage for N Reactor fuel Details on these modifications are provided in the K Basins FSAR, HNF-SD-WM-SAR-062

The KW Reactor was shut down in February 1970 and the KE Reactor one year later, in February 1971 The stored fuel, except for a few loose pieces, was shipped to the 200 East Area for processing The storage basins were then idle but were kept filled with water The area water system was shut down, except for a small portion periodically activated to provide a reservoir of water for sanitary and fire protection systems, process water for other activities, and makeup water for the fuel storage basins

N Reactor spent fuel currently stored in the K Basins was destined for chemical processing in the Plutonium-Uranium Extraction (PUREX) Facility However, in December 1992, DOE directed that the PUREX Facility be shut down and deactivated because it was no longer needed to support the nation's weapons-grade plutonium production
In addition to the storage of N Reactor fuel, a small amount of single pass reactor (SPR) fuel is also stored in the K Basins. The KW Basin has 98 kg, 87 pieces, of unidentified initial $^{235}\text{U}$ enrichment SPR SNF of 1.25 wt% $^{235}\text{U}$ or less encapsulated in canisters. The KE Basin has 396 kg, 138 pieces, of less than or equal to 0.95 wt% $^{235}\text{U}$ fuel stored in 3 canisters and 2 open containers (see K Basins FSAR, HNF-SD-WM-SAR-062). The KW Basin received approximately 3.5 metric tons of uranium of SPR fuel from cleanout of the PUREX Facility in 1996.

The K Basins FSAR covers the handling (including load-in and load-out), encapsulation, storage of N Reactor fuel, and storage of small amounts of SPR fuel in the modified fuel storage basins. During normal operation, there are no releases of significant quantities of radioactive materials to the environment. Contaminated or potentially contaminated building service floor drains within the basins' boundaries have been intercepted and routed to a liquid effluent sump for return to the basins. Unused or unnecessary drains have been plugged and sealed with concrete. An epoxy sealant has been applied to the floor and walls of the KW Basin to further limit leakage. An epoxy sealant was applied to the discharge chute area of the KE Basin after leak repair in 1983. The existing underbasin leakage collection systems composed of an asphalt membrane and a pipeline to a dispersion tile field have been intercepted outside the facility. If leakage occurred contaminated effluents would be collected, routed to a sump and pumped back to the facility or to a radioactive waste holding tank. Leakage from the discharge chute area of the basins is not intercepted by the underbasin leakage collection system. Basin water is passed through an ion exchange medium to remove fission products.

Design basis accidents at the K Basins are discussed in Section 3.4.2 of the K Basins FSAR. Several credible accidents at the K Basins have been determined such as the dropping of heavy loads in the basin, drop of a cask–MCO during cask removal, and drops impacting basin drain lines.

The unmitigated dose from these accidents exceeds both guidelines and limits. There are no mitigated dose consequences because safety-class design features and administrative controls prevent basin leakage from occurring.

Another DBA includes ignition of a fuel element in an MCO basket which exceeds the evaluation guidelines for the onsite receptor at 100 m for an unmitigated dose. Design of the MCO basket stiffback grapple precludes raising the basket above the basin water level and prevents this accident from occurring.

As described in HNF-SD-SNF-SAD-002, the Integrated Water Treatment System (IWTS) filters water from the fuel retrieval system (FRS) and maintains basin water quality for dose minimization and water clarity. The IWTS supplies treated water for fuel removal processes and other uses in the basin.
K Basin DBAs involving the IWTS include (1) fire and explosion hazards and (2) spray releases. Preliminary results indicate that the onsite (100 m, east) unmitigated dose is exceeded. This event has an estimated extremely unlikely frequency of occurrence and potential dose consequences are orders of magnitude lower when credit is taken for the safety-significant designation of the annular filter vessel radiation monitoring system used to mitigate the filter backwash accident scenario.

Preliminary results from a spray release initiated by a rupture in an IWTS pressurized process line indicate the onsite (100 m, east) unmitigated release is also exceeded. This event is considered an unlikely operational accident. Actions taken for mitigating the spray release consequences during filter vessel backwash were evaluated and result in a significantly lower dose.

The radiological dose consequences for the unmitigated hydrogen deflagration are less than the evaluation guidelines. Safety-significant equipment reduces dose consequences by several magnitudes. Mitigated dose consequences for the IWTS spray release are less than the evaluation guidelines.

Additional K Basin DBAs are discussed in Section 3.4.2 of the K Basins FSAR. Mitigated dose consequences of all K Basin DBAs are less than the evaluation guidelines. If CVDF habitability is challenged by K Basin accidents, actions (e.g., notifications, take-cover, emergency shutdown, evacuation) will be performed in accordance with the CVDF emergency plan and emergency response procedures. Facility requirements for operator actions to reach a safe and stable state and permit evacuation are described in Section B4.3.1.4.

**B1 7112 N Reactor** The N Reactor was a 4,000 MW thermal, graphite-moderated, pressurized, light-water-cooled, dual-purpose reactor capable of producing plutonium and, as a by-product, provided steam for the Energy Northwest (formerly the Washington Public Power Supply System) Hanford Generating Project (HGP). The reactor operated from 1963 through 1987. The N Basin received SNF from the reactor and was equipped with the capability to inspect, store and prepare the SNF for shipping. The reactor block was unloaded of fuel and all remaining spent fuel was removed from the N Basin in 1989. A cease preservation directive was received from DOE in 1991.

The N Reactor is currently being maintained in a surveillance and maintenance mode referred to as the "quiescent state." Deactivation of N Reactor and associated facilities, involving removal and disposal of radioactive and hazardous materials has been completed.

Risks at the N Reactor were segmented as to those associated with the N Basin and those associated with the N Reactor complex. The nature of the hazards (quantities, form, and location), their vulnerabilities, and the controls appropriate to maintain the N Basin hazards in an acceptable condition are addressed in BHI-00866, *N Basin Deactivation Project Hazardous Baseline*. This report identifies no N Basin hazards vulnerabilities that require reliance on either...
active mitigative or protective measures adequate facility structural integrity exists to safely control hazards. The report identifies no risks that could impact K Basins or CVDF operation.

Intrusive activities taking place during N Basin deactivation involved the removal and disposition of radioactively contaminated and activated material, basin sludge and basin water. Removal of high-exposure hardware and activated fuel fragments from the basin has been completed. Sludge characterization and removal is nearly complete and removal of basin water has been completed. Removal of basin sludge is currently scheduled to be completed before the SNF movement to the CVDF begins. If the status of the N Reactor changes, potential impacts to the CVDF will be re-evaluated.

**B1 7 1 1 3 Single-Pass Reactors** Eight single-pass water-cooled graphite-moderated reactors were used for plutonium production B, C, D, DR, F, KE, KW and H. The KE and KW Reactors were similar to the other SPRs but had larger reactor blocks in terms of physical size and the number of process tubes and control rods. Operation of the SPRs began with the startup of B Reactor in 1944 and ended with the shutdown of KE in 1971. The risk of these reactors to workers and the public is addressed in WHC-EP-0619 *Risk Management Study for the Retired Hanford Site Facilities Volume 2 Risk Evaluation Work Procedure for the Retired Hanford Site Facilities*. While this report concludes that significant risk and risk management problems exist with these facilities, the risks are associated with worker safety primarily related to the potential for falls and electrical shock. This study considered the radiological hazards associated with the significant amount of $^{60}$Co and $^{14}$C contained in the defueled reactor blocks and associated equipment. It did not give specific attention to the radionuclide inventory in the soils resulting from the following operations:

- Disposal of solid waste
- Disposal of liquid waste
- Leakage and overflow from cooling water retention basin
- Use of cribs and trenches to divert reactor coolant contaminated by fuel failures

The most significant release mechanisms from these soil sources is desorption of the immobile contaminants from the aquifer matrix and subsequent infiltration from the contaminated soils into the underlying groundwater. This groundwater eventually discharges into the Columbia River where it can contaminate sediments and has the potential to cause adverse impacts on local biota with possible food-chain effects on humans offsite (DOE/RL-90-33 and DOE/RL-91-46). More direct access to this groundwater in the vicinity of the reactors will not exist during CVDF operation as institutional controls will remain in place during this relatively short period. Most, if not essentially all of the contamination is now buried beneath the ground surface. These sites are permanently marked as radiation zones with concrete markers and warning signs, and the material is isolated from accessible areas. As such, the greatest risk of direct human exposure is to onsite workers who are involved in collecting environmental samples and conducting remedial activities (DOE/RL-90-33). Neither of these exposure paths represents a risk to SNF Project operation.
The DOE record of decision for disposal of these reactor facilities states that the preferred alternative is to place the reactors in safe storage for up to 75 years and then move the reactor blocks, in one piece, to a burial facility in the 200 West Area. The safe storage activities for the C Reactor were started in 1996 and completed in 1999. The reactor building was removed and the remaining reactor block cocooned (made inaccessible). If the status of the SPR changes (i.e., active decontamination and decommissioning), potential impacts to the CVDF will be re-evaluated.

B1 7 1 2 Hazards to the 100 K Area Facilities from Nonnuclear Hanford Site Facilities

B1 7 1 2 1 Water Treatment Facilities Sodium hypochlorite has replaced the chlorine water treatment at the 183-KE Water Treatment Facility and the 100 N Water Treatment Plant. The new water treatment process no longer presents a potential hazard to 100 K Area facilities operators.

B1 7 1 2 2 Hanford Generating Project The HGP, located adjacent to the N Reactor, was operated by Energy Northwest. The HGP consisted of two 430 Mwe low-pressure turbine generators that received steam from the N Reactor. It also included associated equipment normally found in a steam power station. The HGP condensers and auxiliary cooling systems were supplied by raw water pumped from the Columbia River and discharged back to the river. Operation of the HGP ceased with the shutdown of the N Reactor. Energy Northwest has decided not to repower the HGP from another source. The HGP contains no hazardous materials that represent a risk to SNF Project operation. Eight potential discrete locations of contaminated soils were associated with HGP operation (DOE/RL-91-46). For the same reasons provided above with regard to similar contaminated soils at the SPR sites, these potential contaminated soil locations do not represent a risk to SNF Project operation.

B1 7 1 2 3 200 Area Facilities A number of nonnuclear industrial facilities operating in the 200 Areas pose the potential for accidental fires, explosions, or releases of toxic fumes. These facilities include the Essential Materials Warehouse (Building 275-EA), oil and paint storage buildings, fabrication shops, gas cylinder storage buildings, the spare parts and electrical warehouse, B Plant storage buildings, maintenance facilities, gasoline service stations, and the powerhouse complexes in each area (284-E and 284-W). The most significant inventory of hazardous material in the 200 Areas exists in Building 275-EA. The worst case chemical release postulated involves evaporation of 2,450 kg of 1,1,1-trichloroethane. With the 8 m separation between this warehouse and the 100 K Area facilities, the inventory represents no threat to the CVDF or the K Basins. Additional hazards information from 200 Area facilities is provided in Section A1 7 of the Canister Storage Building FSAR, Annex A.

B1 7 1 3 Hazards to the Cold Vacuum Drying Facility from Nuclear Reactors Three recently shut down reactors, the N Reactor, Fast Flux Test Facility, and the Critical Mass Laboratories, do not pose a threat to the 100 K Area facilities. The N Reactor is undergoing decontamination and decommissioning, the Fast Flux Test Facility is in standby mode and may...
operate again in the future, and the Critical Mass Laboratories in the 200 East Area north of the PUREX Facility are currently being used as tank farm office areas.

The N Reactor was a 4 000 MW dual-purpose pressure tube, light-water cooled, graphite-moderated reactor (UNI-M-90). It is located in the 100 N Area, 2 4 mi from the 100 K Area site. The N Reactor began operating in 1963 producing plutonium for the defense program and steam for electrical power generation. The reactor was shut down in 1987 for safety improvements and then subsequently defueled and placed in cold standby in 1988. It is currently being decontaminated and decommissioned. Current plans are to remove the reactor building structures for the production reactors down to the reactor block and then cocoon the reactor block for 75-year safe storage.

The Fast Flux Test Facility is a 400 MW sodium-cooled mixed-oxide-fueled, fast reactor used for testing breeder reactor fuels, materials, and components. The Fast Flux Test Facility is currently in standby status, with the fuel removed from the core. Sodium is kept circulating in the loops at 400 °F until a decision is made to either restart the reactor for future missions or deactivate and decommission it. The facility is located in the 400 Area, 18 mi from the 100 K Area site.

The only operating nuclear reactor on the Hanford Site is Washington Nuclear Plant 2 (WNP-2) operated by Energy Northwest. The location of the Energy Northwest facilities is shown in Figure B1-1. WNP-2 is an operating commercial nuclear power plant using a boiling-water reactor steam supply system. The design power level was increased to 3,486 MW in 1995 (Docket No 50-397). The reactor was designed by the General Electric Company and is designated as a BWR/5 with a Mark II containment.

According to the requirements of Title 10, Code of Federal Regulations, Part 100 Reactor Site Criteria (10 CFR 100), the following are the maximum allowable doses for WNP-2:

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<th>Location</th>
<th>Duration</th>
<th>Whole body dose</th>
<th>Thyroid</th>
</tr>
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<tr>
<td>Exclusion area boundary</td>
<td>2 hours</td>
<td>250 mSv (25 rem)</td>
<td>3 000 mSv (300 rem)</td>
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<tr>
<td>Low population zone</td>
<td>30 days</td>
<td>250 mSv (25 rem)</td>
<td>3 000 mSv (300 rem)</td>
</tr>
</tbody>
</table>

The exclusion area boundary for WNP-2 is 1.2 mi and the low population zone distance is 3 mi. The K Basins and CVDF are located approximately 17 mi from WNP-2. Using the atmospheric diffusion guidance provided in NRC Regulatory Guide 1 3 Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors to estimate the dose reduction as a function of distance, it was determined that the 2-hour and 30-day doses at the 100 K Area would be reduced by a factor of 30. Review of the site-specific meteorology provided in the WNP-2 FSAR (Docket No 50-397) shows there...
would be no significant reduction for the change in wind direction. The factor of 30 reduction for distance would result in a whole body dose of 8.3 mSv (0.83 rem) and a thyroid dose of 100 mSv (10 rem).

The expected dose that would be received at the 100 K Area site, if a loss of coolant accident occurs, would be significantly less. NRC Regulatory Guide 1.3 requires an assumption that 25% of the radioactive iodine and all of the noble gases are released to the containment. In fact, the emergency core cooling system would prevent most of these releases, as little fuel damage would occur as a result of the loss of coolant accident.

Energy Northwest plans to add an independent spent fuel storage installation on its leased property. The independent spent fuel storage installation would be licensed based on the requirements of 10 CFR 72. According to 10 CFR 72.106, an individual located at the independent spent fuel storage installation's controlled area boundary shall not receive a dose greater than 50 mSv (5 rem). At the 100 K Area site, this would result in a dose not exceeding 1.7 mSv (0.17 rem).

B1 7.1.4 Hazards to the 100 K Area Facilities from Industrial Facilities off the Hanford Site

There are no oil or gas pipelines in the vicinity of the K Basins and CVDF. The nearest major natural gas pipeline to the CVDF site is about 34 mi away. A 20-in gas transmission line of the Northwest Pipeline Corporation is located east and essentially parallel to U.S. Highway 395 between Pasco and Ritzville, Washington. A second pipeline system consisting of parallel 36-in and 42-in lines, owned by Pacific Gas Transmission Company, passes through Wallula, approximately 65 mi from the site (Hosler 1996). These distances eliminate any potential hazard to plant operations from a natural gas fire or explosion due to industrial pipelines.

The nearest petroleum product storage tanks are located 47 mi from the site. These are 23-million-gallon capacity tanks at the Chevron Pipeline Company, and 21-million-gallon capacity tanks at the Tidewater Barge Lines. There are no plans to use a third petroleum storage facility at the Port of Pasco (Hosler 1996).

No other nonnuclear industrial facilities or operations have been identified that may impact 100 K Area facilities operations.

B1 7.1.5 Hazards to the 100 K Area Facilities from Military Facilities

The Yakima Training Center is a subinstallation under the command of Fort Lewis (Tacoma, Washington). Further information is given in the Final Environmental Impact Statement — Ft. Lewis Military Installation (DOA 1979). The southeastern boundary of the Yakima Training Center is located about 12 mi from the 100 K Area site (Figure B1-13). The Yakima Training Center is used for military maneuvers and weapons training and is the only significant military activity in the vicinity of the Hanford Site.

The only weapon currently in use at the Yakima Training Center known to present a potential hazard to the Hanford Site is the Multiple Launch Rocket System (MLRS). With a
range of approximately 16 mi; the MLRS could potentially impact the 100 K Area site. However, the MLRS is only fired from the perimeter of the Yakima Training Center into a centrally located impact zone. The safety fan for the MLRS is shown in Figure B1-13. The MLRS is fired away from the Hanford Site and is only fired with dummy warheads. Given this information, additional safety features and the administrative controls in place at the Yakima Training Center, a weapons accident having an impact on the Hanford Site is very improbable.

A more probable hazard to Hanford Site facilities is a scenario in which a fire starts within the Yakima Training Center boundary and spreads to the Hanford Site. Exploding artillery shells, sparks from tracked vehicles or other machines, and careless smoking by troops might start brush fires that, under adverse meteorological conditions, could spread rapidly beyond the Yakima Training Center boundaries. Range fires were considered with other potential hazards in the hazards analysis and Chapter B3-0 and were not found to adversely impact the CVDF.

B1 7.2 Potential Effects of the Cold Vacuum Drying Facility on Nearby Facilities

Potential CVDF accidents are discussed in Section B3-4.2. CVDF hazard assessments as discussed in Section B15-4.2 will use these potential accidents to help develop the CVDF emergency response plan and procedures, as discussed in Section 15-4 of the SNF Project FSAR and Section B15-4.

The CVDF hazard assessments for emergency planning for the CVDF will characterize the potential consequences to workers, the public, and the environment for each postulated accident. The hazard assessments will also determine the emergency planning zone as well as the emergency class protective actions and the observable indications and criteria (emergency action levels) corresponding to the range of potential accidents.

The hazard assessments will also include information to determine potential impacts to nearby facilities. This information will be made available to nearby facilities for determination of appropriate actions to be included in their emergency response plans and procedures. Information contained in CVDF emergency response plans and procedures will also include notifications evacuation potential operations impacts radiation threats to workers and other information relevant to nearby facilities. Prompt and accurate emergency notifications will be performed in accordance with DOE O 1511 Comprehensive Emergency Management System DOE O 232 1A, Occurrence Reporting and Processing of Operations Information applicable Federal, state or local requirements and special agreements with offsite agencies or tribal governments to mitigate the consequences and to protect the health and safety of workers, the public, and the environment.
B1 8 VALIDITY OF EXISTING ENVIRONMENTAL ANALYSES

No significant discrepancies have been identified between the site characteristic assumptions made in this chapter and those made in the SNF Project Environmental Impact Statement (DOE/EIS-0245F)

B1 9 REFERENCES


10 CFR 100, 1995 "Reactor Site Criteria," Code of Federal Regulations


BHI-00866, 1996 N Basin Deactivation Project Hazardous Baseline, Rev 1, Bechtel Hanford Incorporated Richland, Washington


DOA, 1979 Final Environmental Impact Statement — Ft Lewis Military Installation, U S Department of Army, Headquarters, 9th Infantry Division and Ft. Lewis, with URS Company, Seattle, Washington

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Annex B — Cold Vacuum Drying Facility


WHC-SD-NR-ER-093 1992 *Foundation Studies WNP-1 100 N Site* Rev 0, Westinghouse Hanford Company Richland Washington


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- Hanford Formation
- Ringold Formation
- Saddle Mountains Basalt
- Wanapum Basalt
- Grande Ronde Basalt

Legend for Elevation:
- Feet
- Meters
- MSL = mean sea level
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100-K Area

- Borehole Location (existing well)
- Borehole Location (new well)

Location of Geologic Cross-Section

K-92B  Borehole Number
(Those numbers beginning with "K" have the prefix "199", e.g., 199-K-22)

- Borehole Location (existing well)
- Borehole Location (new well)

Location of Geologic Cross-Section

0  200  400 Meters

2G9607019B 17a
R1 CVDS-1

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- Hog Ranch
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2G96070195 1
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Hz
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CHAPTER B2 0

FACILITY DESCRIPTION
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**LIST OF TERMS**

- **ALARA:** as low as reasonably achievable
- **ARM:** area radiation monitor
- **ATC:** automatic temperature control
- **CAM:** continuous air monitor
- **CSB:** Canister Storage Building
- **CVDF:** Cold Vacuum Drying Facility
- **DOE:** U.S. Department of Energy
- **FSAR:** final safety analysis report
- **HEPA:** high-efficiency particulate air (filter)
- **HVAC:** heating, ventilation, and air conditioning
- **IXM:** ion exchange module
- **MCO:** multi-canister overpack
- **MCS:** monitoring and control system
- **MSLD:** mass spectrometer leak detector
- **NRC:** U.S. Nuclear Regulatory Commission
- **PLC:** programmable logic controller
- **PWC:** process water conditioning
- **RGM:** residual gas monitoring
- **SCHe:** safety-class helium
- **SCIC:** safety-class instrumentation and control
- **SMP:** contaminated water sampling and analysis
- **SNF:** spent nuclear fuel
- **SSC:** structure, system, and component
- **UBC:** Uniform Building Code
- **UPS:** uninterruptible power supply
- **VPS:** vacuum purge system
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B2 0 FACILITY DESCRIPTION

B2 1 INTRODUCTION

The Spent Nuclear Fuel (SNF) Project Cold Vacuum Drying Facility (CVDF) mission is to remove free water from metallic uranium SNF and to prepare the SNF for storage in an inert atmosphere (helium). The SNF will be packaged into water-filled multi-cylinder overpacks (MCOs) and removed from the 100 K Area K East and K West fuel storage basins at the U.S. Department of Energy (DOE) Hanford Site in southeastern Washington State. Removal of free water is necessary to halt water-induced corrosion of exposed uranium surfaces (which results in the generation of hydrogen gas and pressurization of the MCO) and to allow the MCOs and their SNF payloads to be safely transported to the Hanford Site 200 East Area and stored within the SNF Project Canister Storage Building (CSB).

The CVDF is located within a few hundred yards of the 100 K Area fuel storage basins just southwest of the 165KW Power Control Building and the 105KW Reactor Building. The CVDF site is in close proximity to all of the required utilities and is within the 100 K Area inner security boundary. The site area required for the facility and vehicle circulation is approximately 20 acres. Access is provided by the main entrance to the 100 K Area using existing roadways.

Removal of free water from the MCOs reduces the potential for fuel-water corrosion reactions that could lead to MCO overpressurization at the CSB. The cold vacuum drying process involves the draining of bulk water from the MCO and subsequent vacuum drying to remove remaining bulk water and sufficient portions of the hydration water. The MCO is evacuated and backfilled with an inert gas (helium). The MCO is then sealed and leak tested before transport to the CSB within a sealed transportation cask.

The CVDF provides the required process systems, supporting equipment, and facilities needed to support the CVDF mission. The CVDF also contains equipment to prepare process water removed from the MCO for transport to the K West Basin and to package solid waste generated by the cold vacuum drying process for disposal. Four process bays contain process equipment, with an unequipped fifth bay.

The MCO remains within a shipping cask on a trailer during all CVDF activities. The trailer design includes a platform that serves as a work station for CVDF operators and provides direct access to cask–MCO process interface connections on the top of the MCO. Access to the lower port of the cask is available through a hole in the lower stabilizer of the trailer.

This chapter provides facility and operating descriptions for the CVDF to support assumptions used in the hazard and accident analyses. These descriptions focus on all major facility features necessary to understand the hazard and accident analyses, not just safety structures, systems, and components (SSCs). The level of detail used in this chapter is based on the significance of the preventive and mitigative features identified and the degree of complexity.
of the systems described. The descriptions provide an understanding of the facility and processes in sufficient detail that a general understanding of the facility can be achieved without extensive consultation of the references cited.

Chapter B30 describes the hazards and accident analyses and safety classifications of the SSCs for the CVDF. Chapter B40 describes the safety-class and safety-significant SSCs in greater detail and also identifies safety controls (the technical safety requirements).

B2 2 REQUIREMENTS

This section lists the requirements that establish the safety basis of the CVDF. The intent is to provide only the requirements that are specific for this chapter and pertinent to the safety basis. The basic process safety and quality assurance requirements, criteria, and data for the CVDF systems and facility are provided in HNF-SD-SNF-DRD-002, *Spent Nuclear Fuel Project Cold Vacuum Drying Facility Design Requirements*. HNF-SD-SNF-DRD-002 also establishes the operational readiness documentation required for testing, operation, maintenance, decontamination, and decommissioning. Specific codes, standards, and requirements applicable to the CVDF are defined in HNF-SD-SNF-RD-001, *Spent Nuclear Fuel Project Standards/Requirements Identification Document*. The specific codes, standards, and requirements applicable to the MCO are identified in HNF-SD-SNF-SARR-005, *Multi-Canister Overpack Topical Report*. Likewise, the specific codes, standards, and requirements applicable to the transport cask are identified in HNF-SD-TP-SARP-017, *Safety Analysis Report for Packaging Onsite Multi-Canister Overpack Cask*.

The design of the CVDF is governed by federal regulations, DOE directives (orders) and standards, Washington State regulations, Hanford Site standards, and various industry standards and guidelines. WHC-SD-SNF-SD-020, *K Basin Spent Nuclear Fuel Cold Vacuum Drying Facility Functions and Requirements* summarizes the requirements of the CVDF and identifies federal, state, and local regulations and laws that are applicable to the CVDF. HNF-SD-SNF-DRD-002 provides functional design and performance requirements for the CVDF.

The following requirements are applicable to the safety basis for the facility:

- **Title 10 Code of Federal Regulations Part 830, Nuclear Safety Management**. Section 830.120, "Quality Assurance Requirements" (10 CFR 830.120). This rule requires that a sufficient quality assurance program be in place.

- **Title 10 Code of Federal Regulations Part 835, "Occupational Radiation Protection"** (10 CFR 835). This rule provides requirements for radiation protection programs.

- DOE Order 5480 23, *Nuclear Safety Analysis Reports* This order provides nuclear safety analysis report content requirements.

- DOE Order 5480 24, *Nuclear Criticality Safety* This order requires that the CVDF design maintain criticality safety mandated spacing requirements during all design basis accidents. The design also incorporates applicable and appropriate results, requirements, and recommendations of HNF-2151, *Criticality Safety Evaluation Report for Multi-Canister Overpack Loading and Handling at the K Basins*.

- DOE Order 5480 28, *Natural Phenomena Hazards Mitigation* This order is used to define design requirements for seismic events and straight wind.

- DOE Order 6430 1A, *General Design Criteria* This order for nonreactor nuclear facilities presents the main reference standards and guides for facility and system design. Division 13, "Special Facilities," Section 1300, "General Requirements," provides overall general design requirements applicable to all SSCs. The contractor will maintain the design configuration, including compliance to DOE Order 6430 1A, throughout facility operations.

- DOE-STD-1020-94, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities* This standard provides the methodology for design of the CVDF for natural phenomena hazard events and is implemented via WHC-SD-SNF-DB-010, *Cold Vacuum Drying System Natural Phenomena Hazards*.

- DOE-STD-1021-93, *Natural Phenomena Hazards Performance Categorization Guidelines for Structures Systems and Components* This standard is used to define the specific performance category for SSCs.

- DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480 23 Nuclear Safety Analysis Reports* This standard is used to determine hazard categories for nuclear facilities.

In this regard, the standard provides more detailed information on the performance of accident analyses for hazard category 2 and 3 facilities. The standard also establishes additional requirements for the establishment of defense in depth and the identification of safety-significant SSCs.

In Letter 95-SFD-167, *Implementation of K Basins Spent Nuclear Fuel Project (SNFP) Regulatory Policy* (Sellers 1995), DOE established the requirement for new SNF Project facilities to achieve "nuclear safety equivalency" to comparable U.S. Nuclear Regulatory Commission (NRC)-licensed facilities. The SNF Project identified the NRC requirements that were needed in addition to existing and applicable DOE requirements to establish nuclear safety equivalency. These NRC requirements and the process used to identify them are documented in HNF-SD-SNF-DB-003 *Spent Nuclear Fuel Project Path Forward Additional NRC Requirements* and in WHC-SD-SNF-DB-010. An assessment of CVDF compliance with SNF Project NRC equivalency criteria is documented in Letter FDH-9950119 *Compliance with DOE 6430 1A and Nuclear Regulatory Commission Equivalency Requirements in Cold Vacuum Drying Project* (Williams 1999a). An update to the assessment is documented in Letter FDH-9953137A R3 *Contract Number DE-AC06-96RL13200 – U.S. Department of Energy Richland Operations Office Review Comments for DOE Order 6430 1A and U.S. Nuclear Regulatory Commission Equivalency Requirements for the Cold Vacuum Drying Facility* (Williams 1999b). Table B4-4 provides a summary of compliance with the additional NRC equivalency items.

**B2.3 FACILITY OVERVIEW**

This section includes a brief overview of the CVDF configuration and the basic processes performed therein. The CVDF provides a facility for draining the water from and drying the SNF-filled MCOs after they have been removed from the K Basins. Removal of free water using a cold vacuum drying process from the MCOs will prevent MCO overpressurization during the projected storage period at the CSB (HNF-SD-SNF-T1-040). The CVDF is designed to be a temporary facility. Removing the entire K Basins inventory of SNF is expected to take two to five years. The design life for the facility and equipment is a minimum of five years. This design life establishes the basis for procurement of equipment and necessary spares. With selective replacement of equipment, the operation of the facility can be extended as future needs dictate.

**B2.3.1 Facility Description**

The CVDF is designed to DOE Order 6430 1A. Furthermore, the process building is designated a nonreactor nuclear facility under Section 1300 "Special Facilities." The administrative building is a nonnuclear facility rated for office/business use in accordance with the *Uniform Building Code* (UBC) (ICBO 1994). The building code requirements are met as defined in the UBC (ICBO 1994) for a Group H-7 occupancy for the process bay and support.
Annex B — Cold Vacuum Drying Facility

areas and a Group B occupancy for the administrative building. See Figure B2-1 for the layout of the four main areas of the CVDF and Figure B2-2 for a side view of the facility.

B2 3 2 Cold Vacuum Drying Facility Hazards

A hazard category classification of the CVDF is presented in Chapter B3 with the assessment that the CVDF is a hazard category 2 nuclear facility. This classification was determined in accordance with DOE-STD-1027-92.

The CVDF hazard analysis is documented in HNF-SD-SNF-HIE-004, Cold Vacuum Drying Facility Hazard Analysis Report. The main inventory of hazardous material in the CVDF is the radiological content of the MCOs. The CVDF does not house chemical process systems.

Other hazardous materials identified in the MCO include pyrophoric metals and hydrides, oxidizers, and hydrogen. In addition, hazardous materials in the process bays and surrounding areas include diesel fuel and other combustible materials. The specific characteristics of those hazards are identified in detail in HNF-SD-SNF-HIE-004. The CVDF fire hazard analysis (SNF-4268) concluded that no release of radioactive material to the environment from the cask and MCO would result from any of the CVDF fire scenarios. As described in Section B3.4.2.3.5, a release to the environment from a flammable explosion in a high-efficiency particulate air (HEPA) filter is maintained below guidelines by a technical safety requirement based on the amount of radionuclides allowed to accumulate on the filter before changeout.

When fully loaded, each MCO houses five (Mark IV fuel ≤ 0.95 wt% \(^{235}\)U enrichment) or six (Mark IA fuel >0.95 wt% \(^{235}\)U enrichment) baskets, each basket containing prescribed amounts of SNF and incidental fuel corrosion products that have been retrieved from the K Basins. See Figure B2-3 for a sketch of a typical MCO and cask package. The following paragraphs contain physical descriptions of the fuel currently in the K Basins, give a fuel burnup summary, and provide details of the chemical and radionuclide inventories of SNF.

HNF-SD-SNF-SARR-005 identifies the restrictions and limitations on fuel contents of the MCOs. A summary discussion of MCO contents is provided in Section B2.5.2.

N Reactor fuel to be processed at the CVDF is Mark IA, Mark IB, Mark IC, Mark IV or Mark IVB fuel. There is, however, a small amount of single-pass reactor fuel currently in the K Basins. The single-pass reactor fuel drying is not covered in these analyses. An addendum to the SNF Project Final Safety Analysis Report (FSAR) will be prepared to address processing of this type of fuel prior to its being loaded into an MCO, processed at the CVDF, and stored in the CSB.

N Reactor fuel assemblies consist of two concentric tubes made of uranium metal coextruded into Zircaloy-2 cladding. The several basic types of fuel assemblies are differentiated by their uranium enrichment. Mark IV fuel assemblies have a preirradiation enrichment of...
0 947 wt% $^{235}\text{U}$ in both tubes and an average uranium weight of 50 lb. The Mark IV assemblies have an outside diameter of 2.42 in and lengths of 17.4, 23.2, 24.6, or 26.1 in. Mark IA fuel assemblies have a preirradiation enrichment of 1.25 wt% $^{235}\text{U}$ in the outer element and 0.947 wt% $^{235}\text{U}$ in the inner element. The Mark IA assemblies have an average uranium weight of 35.9 lb. Mark IA fuel assemblies have an outside diameter of 2.40 in and lengths of 14.9, 19.6, 20.9, or 26.1 in. Mark IB and Mark IVB fuel assemblies have preirradiation enrichments of 0.71 wt% $^{235}\text{U}$ and Mark IC fuel assemblies have a preirradiation enrichment of 0.95 wt% $^{235}\text{U}$. There are no special handling requirements for these fuel types. The Mark IB and Mark IC fuel assemblies are placed in Mark IV baskets. The Mark IVB fuel has the same dimensions and weights as the 26.1-in-long Mark IV fuel. A diagram of a typical Mark IV fuel assembly is provided in Figure B2-4.

Exposure level or fuel burnup and time since discharge determine the radionuclide content of a fuel assembly or group of assemblies. The K Basins inventory of N Reactor fuel is composed of elements that experienced a range of exposure levels. The exposure levels ranged from unburned fuel (0 MWd) to approximately 6,000 MWd per metric ton of uranium. This level is related to the weight fraction of the isotope $^{240}\text{Pu}$ that exists within the total quantity of plutonium in a particular fuel element. The $^{240}\text{Pu}$ content is commonly used to indicate fuel burnup and ranges from approximately 0 wt% $^{240}\text{Pu}$ up to 16.72 wt% $^{240}\text{Pu}$ for the N Reactor fuel in inventory.

Accountability records have been used as the basis for estimating the radionuclide content of N Reactor fuel. The accountability record run data include discharge date, fuel type, $^{240}\text{Pu}$ content, and other information for 497 groups of fuel elements (the groups are also known as keys). Each group or key includes elements of the same type with the same burnup, that were discharged from the reactor at the same time. The mass of uranium associated with each group varies from 7 kg to 67.4 metric tons of uranium. The accountability database that forms the inventory basis is shown in Appendix A of HNF-SD-SNF-TI-009. The Basin Material Design Basis Feed Description for Spent Nuclear Fuel Project Facilities.

The chemical content of the fuel on a preirradiation basis is derived by applying the reported preirradiation concentration range to the total uranium (4,630,000 lb), zirconium (320,000 lb), and braze fill (6,600 lb) in N Reactor fuel. In the post-irradiation fuel, a fraction of the $^{235}\text{U}$ has been fissioned, generating fission products. A small quantity of the $^{238}\text{U}$ has been converted to plutonium, and other elements in the fuel, cladding, and fuel element hardware have been activated by exposure to the neutron bombardment.

B2.3.3 Cold Vacuum Drying Facility Operations Summary

CVDF operations start with the receipt of the cask-MCO trailer at a process bay. The shipping documentation package is verified. Operators raise the process bay door to allow backing the tractor-trailer into a predetermined position in the bay. The tractor is driven out of
the process bay and the bay door is closed. Bay confinement is then reestablished. The security system is activated for the bay.

During transport to the CVDF, the MCO is vented to the cask headspace. Internal cask pressure is expected to increase as a result of hydrogen gas generation, and temperature is expected to increase as a result of radioactive decay heat, solar heating, and water-uranium corrosion reactions. Cask venting is performed by means of special venting hardware and flex lines connected to a cask lid port and the CVDF process bay local exhaust heating, ventilation, and air conditioning (HVAC) and process vent system. After venting, the cask lid is removed by a process bay overhead crane, the process hood seal ring assembly is installed onto the cask, and the MCO is prepared for process operations. See Section B2.5.1 for an overview of the cold vacuum drying process.

MCO processing is performed per operating procedures at a controlled water temperature in the cask-MCO annulus. Normal processing includes bulk water removal, helium purging, evacuation with and without helium purge, initial pressure rebound testing, extended operation under vacuum at the base pressure of the system, final pressure rebound testing, integrated leak testing of the MCO mechanical seals, and backfilling with helium. Normal processing is conducted between 40 °C and 50 °C. There are minimal manual operator actions in the process sequences. Field operator actions such as connecting the MCO process connectors, the deionization/helium rinse, blowdown after draining, and tempered water connections are required. Control room operators are responsible for acknowledging alarms and directing the monitoring and control system (MCS) and safety-class instrumentation and control (SCIC) system when to initiate a processing sequence. Valve state changes, water temperature control, and other process parameter changes are performed by the MCS.

There are two normal operator interfaces, the MCS computers and the SCIC system mode switch, both located in the control room (room 107). The SCIC system mode switch defines the MCO process mode and directly feeds this information into the MCS. While most sequences are held for operator permissions, some sequences occur based solely on SCIC system mode (e.g., keeping hydrogen concentration low in the MCO and process water conditioning [PWC] system receiving tanks).

Following the cold vacuum drying process and MCO testing, the cask-MCO trailer is prepared for shipment to the CSB. This operation is as follows:

- The cask-MCO is brought to approximately room temperature
- The MCO is pressurized with helium, sealed, and leak tested
- The cask annulus is drained and dried and the cask lid is reinstalled
- The cask-MCO trailer is reconnected to the tractor and released for shipment to the CSB
The CVDF can process up to four MCOs at any one time (one for each active process bay). However, the PWC system is designed to service one MCO at a time, so priority determination and scheduling are important.

### B2 3 4 Cold Vacuum Drying Facility Confinement

Confinement provides retention of radiological and chemical materials in specified areas. Confinement features in use at the CVDF can be divided into two categories: (1) process confinement for the SNF, and (2) ventilation confinement of gases in the CVDF.

#### B2 3 4 1 Spent Nuclear Fuel Confinement

During transportation to the CVDF, confinement for the SNF consists of (1) the MCO and (2) the transportation cask. At the CVDF, the tractor-trailer is decoupled, leaving the trailer with the loaded cask-MCO in one of the four processing bays. At this time, the cask lid is removed temporarily breaching SNF confinement for the water-filled MCO and a process hood is installed over the cask-MCO to reestablish secondary confinement via ventilation control.

#### B2 3 4 2 Cold Vacuum Drying Facility Confinement

Confinement in the CVDF is presented in two categories: primary and secondary.

##### B2 3 4 2 1 Primary Confinement

Primary confinement acts as a barrier separating the SNF material contained in the MCO and the process piping from the surrounding secondary confinement systems. The primary confinement of airborne and liquid effluents from the SNF is provided by the MCO (see Section B2 5 2 for a discussion of the design of the MCO and transportation cask) and isolation piping.

##### B2 3 4 2 2 Secondary Confinement

Secondary confinement controls movement of material released from the primary confinement system. During MCO processing, normal vent paths and exhausts are vented to secondary confinement. Some accident scenarios release contamination within the secondary confinement, which is captured and vented through a HEPA-filtered exhaust system. Secondary confinement consists of the process bay local exhaust HVAC and process vent system and the CVDF structure in conjunction with the process general supply/exhaust HVAC system. The process bay local exhaust HVAC and process vent system handles all potentially contaminated air flows from the process hood, process vents from the vacuum purge system (VPS) and tempered water (annulus) system cask venting the safety-class helium (SCHe) system and the VPS and PWC system piping beyond the isolation valves. During normal operations, the process general supply/exhaust HVAC system maintains a negative differential pressure within potentially contaminated areas of the CVDF (process bays, transfer corridor and associated process support rooms, the mechanical room, and the process water tank room). Differential pressure indicators are provided throughout the facility. This system also supplies conditioned air to and exhausts from the process support areas and the process water tank room. Each process bay has a HEPA-filtered recirculation system that is controlled individually to maintain temperature control and air quality with outside makeup air. The spare
process bay (bay 1) is not normally ventilated but has an installed connection to the process general supply/exhaust HVAC system

See Section B2 6 for a more detailed discussion of facility confinement and building HVAC

B2 3.5 Cold Vacuum Drying Facility Systems

Table B2-1 contains a list of CVDF systems. The system descriptions in this section are presented in the same order as they are given in the table. Table B2-1 identifies the Chapter B2 0 section that addresses the system, identifies the system number, and provides the system abbreviation.

B2 4 FACILITY STRUCTURE

The CVDF is approximately 230 ft in length, 80 ft in width, and 35 ft in height. The CVDF has the following four main areas:

- Process bay area
- Process support area
- Process water tank room
- Administrative building

The process bay area (60 ft wide by 150 ft long by 35 ft high) contains five process bays (one is a spare bay without services). The process support area (20 ft by 150 ft) includes a transfer corridor and adjacent rooms, along with a second-floor mechanical room, and is constructed as a two-story steel frame building with an exterior wall of metal siding. The process water tank room (20 ft by 40 ft) adjoins the north wall of process bay 1. It is constructed as a single-story, steel-frame building with 10-in.-thick exterior walls of precast concrete panels. The administrative building, adjacent to process bay 5, is a single-story, preengineered metal building with an exterior wall of insulated metal panels. All roof decks are metal. The CVDF exhaust stack (48 ft high and 30 in. in diameter) is located 17 ft from the west wall of the process support area.

Each process bay (excluding the spare bay) contains a second-level mezzanine. The process bay area design consists of a steel frame with attached concrete panels to facilitate decontamination and demolition.
### Table B2-1  Cold Vacuum Drying Facility Systems  (3 sheets)

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The construction of the CVDF is in accordance with DOE Order 6430 1A, with egress requirements conforming to NFPA 101 Life Safety Code The process bay area is designated a nonreactor nuclear facility under DOE Order 6430 1A Section 1300 "Special Facilities " The administrative building is a nonnuclear facility rated for office and business use in accordance with the UBC (ICBO 1994) The building code requirements are met as defined in the UBC (ICBO 1994) for a Group H-7 occupancy for the process bay and process support areas and a Group B occupancy for the administrative building The exterior walls of the CVDF are prestressed concrete panels for the process bay area and process water tank room and insulated metal panels for the process support area and the administrative building

This section provides an overview of the facility and auxiliary structures, including construction details (e.g., basic floor plans, equipment layout, construction materials, controlling dimensions and dimensions significant to the hazard and accident analysis activity). This information supports an overall understanding of the facility structures and the general arrangement of the facility as it pertains to hazard and accident analyses. See Figure B2-5 for the CVDF first-floor layout, Figure B2-6 for the CVDF second-floor layout, Figure B2-7 for architectural exterior elevations, and Figure B2-8 for the process bay sections. The following subsections describe the process bay area (the process bays), the process support area (transfer corridor, mechanical equipment room), the process water tank room, and the administrative building (lunch/conference room, offices, control room, electrical and mechanical rooms, change rooms).

### B2 4.1 Process Bay Area

Each of the four active process bays (process bays 2 through 5) contained within the process bay area is designed to enclose a cask-MCO trailer without the tractor attached and to provide the operational space necessary to meet the functions of the cold vacuum drying process. Process bay area design provides radiological separation and shielding between process bays and ventilation confinement within each process bay. The spare process bay (bay 1) does not contain any process equipment and does not have a mezzanine. The bay does contain equipment and

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HVAC = heating, ventilation, and air conditioning
MSLD = mass spectrometer leak detector
N/A = not applicable
connections for transferring processed water from the PWC system to a transport vehicle to return to the K Basins.

The process bay area design is a steel frame and prestressed concrete panel system that has a process bay width of approximately 30 ft and a nominal length of 60 ft. The height of the process bays is nominally 32 ft, which is dictated by the physical and functional requirements for all of the operations necessary. The exterior walls are 10-in-thick precast concrete panels. The interior walls are 8-in concrete panels up to a height of 23 ft. The panels are precast and prestressed, and the vertical joints are caulked. The remaining 9 to 12 ft above the interior precast concrete wall to the sloped roof is constructed of gypsum wallboard coped around the beams and sealed. The floor is sloped to a sump. An overhead telescoping metal door is provided for trailer entry into each process bay. Bay 1 has a roll-up door. A U-shaped mezzanine is provided in each process bay to support the HVAC system, to provide access to the process hood, and to provide support for activities at the top of the MCO. The mezzanine floor is constructed of steel plates. The roof is built-up membrane roof over rigid insulation adhered to a steel plate deck.

Each process bay includes an overhead bridge crane with a capacity of 2 tons. The crane is designed in accordance with the criteria included in CMAA 74, Specifications for Top Running and Under Running Single Girder Electric Overhead Traveling Cranes Utilizing Under Running Trolley Hoist. The bridge, trolley, and hoist are driven by electric motors and are capable of operation from the mezzanine level.

Access to the process bay area is via the process support area. Individual process bay access and egress control is through a vestibule with personnel doors to the transfer corridor and the process support rooms.

Each process bay provides ground floor space for the following:

- Enclosing a cask trailer without the tractor attached
- Personnel circulation and functional space around the cask trailer
- Process equipment skid including vacuum drying and tempered water equipment
- Access to the working level of the cask trailer
- Bridge crane control to remove the cask lid, install the process hood, and perform maintenance on equipment
- A supply cabinet and storage locker
- Electrical control panels
Access to the working level of the cask is accomplished using the mezzanine level, which has space for the following:

- Access to the working level of the cask trailer for process connections to the MCO
- HVAC equipment
- Process hood

## B2.4.2 Process Support Area

The process support area includes the transfer corridor and adjacent rooms on the first floor and the mechanical room on the second floor.

The two-story process support area is 28 ft high and 20 ft wide along the entire 150-ft length of the west side of the process bay area. Ventilation for both floors of the process support area is provided by the process general supply/exhaust HVAC system (see Section B2.6.2.5).

The process support area is built on a concrete foundation and is separated from the process bay area by concrete panel walls. The process support area structure is attached to the process bay area structure. The other interior walls in this area are gypsum wallboard on a metal frame construction. The exterior wall of the process support area is insulated metal siding. The second floor is 4-in-thick concrete on a metal deck supported by the structural steel frame. The standing beam fiberglass-batt insulated metal roof is supported by the structural steel frame.

The transfer corridor (corridor 116) runs the length of the process bay area along the west end of each process bay and ends at the process water tank room. This corridor is contiguous with the rooms at the south end of the administrative building. Individual process bay access control is provided through a vestibule specific to each process bay. In addition to the process bay access vestibules, corridor 116 provides access to process support area rooms 117, 120, 123, 126, and 129.

The process bay access vestibules (rooms 118, 121, 124, 127, and 130) function as ventilation boundaries between the controlled process bay and the uncontrolled transfer corridor. They are approximately 19 ft by 12 ft.

A single personnel decontamination room (room 117) serves all process bays via the transfer corridor. The personnel decontamination room provides clear standing space for the decontamination process, storage of decontamination materials and detection equipment, and one decontamination shower.
A swipe-count room (room 123) is provided to analyze and store samples taken from process bays or areas that have the potential for contamination. This room provides desk space for preparing testing reports, file space for storage of records, space for counting equipment, and wall space for mounting gas bottle equipment.

A deionized water equipment and air compressor room (room 120), with approximately 120 ft² of enclosed floor space, is located in the process support area.

A security system equipment room (room 126), with approximately 120 ft² of enclosed floor space, is located in the process support area.

The mechanical equipment room (room 207) (20 ft by 150 ft) is located on the second floor of the process support area above the transfer corridor, process bay change rooms, and other support rooms. Building ventilation equipment for the process bay local exhaust HVAC and process vent system and the process general supply/exhaust HVAC system are located in this room, along with chilled water system equipment.

**B2.4.3 Process Water Tank Room**

Water removed from an MCO has the potential to be contaminated, therefore, the water will be stored in a tank located in the isolated process water tank room with controlled access until it is transferred to K West Basin. The process water tank room (room 132) is located directly adjacent to the CVDF process bay area and process support area. This room is located adjacent to process bay 1. The process water tank room is approximately 20 ft wide, 40 ft long, and 13 ft high. The room is built on a concrete foundation and is separated from the process bay area by concrete interior walls. The process water tank room structure is attached to the process bay area structure. The exterior wall of the process water tank room is precast concrete. The process water tank room floor has an elevation of 2 ft below the rest of the CVDF and is designed with elevated doorways to contain liquid leaks.

**B2.4.4 Administrative Building**

The administrative building is the normal CVDF personnel entry point and provides space for a lunch and conference room, quality assurance and process engineer office, shift manager's office, radiological control technician and radiation monitoring office, control room, electrical and telecommunications room, fire riser and mechanical room, men's and women's rest rooms, men's and women's change rooms, and access and egress control and control monitoring of the process bays.

The administrative building is approximately 50 ft wide by 60 ft long by 12 ft high. This area is separated from the adjacent process support area by a 2-hour fire-rated wall. The adjacent process bay area wall is a precast panel that has fire resistance equivalent to a 2-hour fire-rated
The interior walls of the administrative building are gypsum wallboard on a metal frame. The exterior walls are standard (butler-rib type) insulated metal siding and the roof over this area is also a standing seam, batt-insulated, metal roof.

The lunch and conference room is room 105. A quality assurance and process engineering office is in room 104, and the shift manager's office is room 103.

The radiological control technician and radiation monitoring office (room 106) has space for associated instrumentation equipment, storage space for radiological control equipment, and wall space for mounting gas bottle equipment.

The control room (room 107) has space for process bay control functions including computer monitoring stations, MCS programmable logic controller (PLC) cabinet space, SCIC system annunciator and mode control panels, security monitors (two) and duress alarm panel, and HVAC control computer.

The electrical and telecommunication room (room 108) provides space for electrical and telecommunication systems and equipment.

The fire riser and mechanical room (room 110) provides space for a fire protection system alarm check valve riser and mechanical accessories and other support equipment.

Separate change room facilities for men and women employees include rest rooms, shower facilities, storage lockers and benches, and space for supplies storage. These are rooms 111, 112, 113, and 114.

**B2 4 5 Cold Vacuum Drying Facility Design Basis**

As discussed in Chapters B3 0 and B4 0, the CVDF SSCs are classified as either safety class, safety significant, or general service. The CVDF building structures are classified as general service and designed to appropriate natural phenomena hazard loads and conditions as described in WHC-SD-SNF-DB-010.

The CVDF building structures are assigned to natural phenomena hazard performance categories 1, 2, or 3 as defined in DOE-STD-1020-94. The process bay area is performance category 3 to protect the safety-class components located in the process bays. The process support area and the process water tank room are performance category 2. The administrative building is performance category 1. The control room is not required to be manned after natural phenomena events and therefore can be designed to the lower performance category.

The design basis natural phenomena loads and conditions applied to the CVDF performance category 3 structural design are tabulated in Chapter B1 0 and in Table 1 of WHC-SD-SNF-DB-010. This information is duplicated in Table B2-2.
<table>
<thead>
<tr>
<th>Hazard</th>
<th>Load</th>
<th>Application documents</th>
</tr>
</thead>
<tbody>
<tr>
<td>Seismic</td>
<td>Performance category 3 (0.26 g) equal hazard response spectra</td>
<td>DOE Order 5480 28</td>
</tr>
<tr>
<td></td>
<td></td>
<td>DOE STD 1020 94</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Table 2 and Figure 1</td>
</tr>
<tr>
<td>Straight wind</td>
<td>80 m/s, fastest mile at 30 ft</td>
<td>ASCE 7 95</td>
</tr>
<tr>
<td></td>
<td></td>
<td>DOE STD 1020 94 (including missiles)</td>
</tr>
<tr>
<td>Tornado</td>
<td>Wind speeds</td>
<td>NRC Standard Review Plan</td>
</tr>
<tr>
<td></td>
<td>322 km/h (200 mph) total</td>
<td>3 3 2 Tornado Loading</td>
</tr>
<tr>
<td></td>
<td>257 km/h (160 mph) rotational</td>
<td></td>
</tr>
<tr>
<td></td>
<td>64 km/h (40 mph) translational</td>
<td></td>
</tr>
<tr>
<td>Volcanic ash</td>
<td>24 lb/ft² ground ash load</td>
<td>NRC Standard Review Plan*</td>
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<tr>
<td></td>
<td></td>
<td>3 8 4 Other Seismic Category Structures</td>
</tr>
<tr>
<td>Flooding</td>
<td>Columbia River 460 ft above mean sea level</td>
<td>ANSI/ANS 2 8 1992</td>
</tr>
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<td></td>
<td>Site drainage basin 7 4 in. for 6 hour probable maximum precipitation</td>
<td>NRC Standard Review Plan</td>
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<tr>
<td></td>
<td>Site drainage 9 2 in. for 6 hour probable maximum precipitation</td>
<td>2 4 2 Floods</td>
</tr>
<tr>
<td>Lightning</td>
<td>Lightning protection shall be provided for facility as required by the</td>
<td>NFPA 780</td>
</tr>
<tr>
<td></td>
<td>code</td>
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<tr>
<td>Snow</td>
<td>20 lb/ft² ground load</td>
<td>ASCE 7 95</td>
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</table>

Note: The wind speed criteria are based on recent revision of the Washington Nuclear Plant 2 tornado criteria by the U.S. Nuclear Regulatory Commission.


*DOE Order 5480 28 Natural Phenomena Hazards Mitigation U.S. Department of Energy, Washington, D.C.
*NFPA 780 1995 Lightning Protection Systems National Fire Protection Association, Quincy, Massachusetts
Performance category 1 and performance category 2 criteria also are defined in
WHC-SD-SNF-DB-010. Seismic design for performance categories 1 and 2 uses UBC
(ICBO 1994) zone 2B. Standard occupancy rules are used for performance category 1
facilities and essential facilities rules for performance category 2 facilities. The design
basis wind speed for performance categories 1 and 2 is 70 m/h. Volcanic ashfall is not
considered for performance categories 1 and 2. The design basis snow load for performance
categories 1 and 2 is 20 lb/ft².

The CVDF structure provides protection from seismic events, high winds, and tornado
winds. The process bay is performance category 3 enclosures for seismic and wind
protection. The process support area and the process water tank room are performance
category 2. Equipment within these enclosures is qualified to the appropriate performance
category requirements to not fail and impact the safety equipment required to accomplish
and maintain a safe shutdown of the process.

There are distinct structural boundaries between performance category 2 and performance
category 3 structural interfaces. Performance category 3 SSCs are isolated from the effects
of other SSC failures such that the performance category 3 safety functions can still be
accomplished under design basis conditions. The performance category 3 process bay area is
isolated from the performance category 2 process support area and the performance
category 2 process water tank room by the design of the connecting walls and structural
attachments.

Seismic design for performance category 3 projects follow DOE Order 5480.28 and
Chapter 2 and Appendix C of supporting standard DOE-STD-1020-94. In-structure response
spectra were generated for SSCs supported by the structure. Performance category 1 and 2
SSCs use the UBC (ICBO 1994) zone 2B, as discussed in DOE-STD-1020-94. Standard occupancy
rules are used for performance category 1 and essential facilities rules for performance
category 2.

There is no operating basis earthquake for the CVDF. The administrative building, which
is performance category 1, is structurally decoupled from the process bay area and is
designed to fail away from the process bay area during a performance category 3 event.

The members and attachments of the administrative building are not strong enough to
withstand performance category 3 forces. They are designed to be stronger against movement
in the north direction than in the south direction so when they fail it is to the south away
from the process bay area.

Straight wind load and missile load criteria are given in Chapter 10 of the SNF Project
FSAR and in Chapter 30 of WHC-SD-SNF-DB-010. Tornado missiles have been eliminated
from the CVDF requirements per WHC-SD-SNF-DB-010. The CVDF performance category 3
structures have been designed for tornado hazards in accordance with NUREG-0800: Standard
Review Plan Section 3.3.2 "Tornado Loading." The performance category 3 structures have
been designed to comply with these criteria as a design basis and therefore will be able to
withstand design basis seismic, wind, and tornado events.

Design basis accident loads, response spectra, and design requirements for ashfall are given
in Chapter 10 of the SNF Project FSAR and in Section 2.2 of WHC-SD-SNF-DB-010.
The design basis river flood level, assuming a 25% breach of Grand Coulee Dam, is 460 ft. The elevation of the CVDF is 476 ft above mean sea level. The CVDF structures are designed to resist earth and groundwater loads in accordance with DOE Order 6430 1A and DOE-STD-1020-94 as administered using the guidance of HNF-PRO-097, Engineering Design and Evaluation. Dynamic earth pressures were considered in the design of performance category 3 below-grade structures. The CVDF structures are designed for both dead loads and live loads in accordance with DOE Order 6430 1A and DOE-STD-1020-94 as administered by the guidance of HNF-PRO-097. The snow-load criterion, as specified in WHC-SD-SNF-DB-010, is the criterion given in Chapter 8 of ASCE-7-95, Minimum Design Loads for Buildings and Other Structures.

Neither DOE nor NRC have developed specific requirements for dealing with lightning. NFPA 780, Lightning Protection Systems, is recommended as guidance by DOE and NRC and has been applied for lightning protection for the CVDF.

The paved areas, concrete floor, and parking slabs of the CVDF are capable of supporting a tractor-trailer with an approximate weight of 105,000 lb. All concrete design complies with American Concrete Institute requirements as defined in ACI-301, Specifications for Structural Concrete for Buildings. Except for the process bays, floors throughout the facility have finishes appropriate to an area's function and flexibility requirements. In the process bays, concrete floors have 100% epoxy-coated surfaces that can be easily decontaminated.

The CVDF location was chosen based on a site selection evaluation that was done to determine the optimal site on which to build (Swenson 1996). The CVDF structural design was selected to facilitate demolition of the facility and recycling of the structural materials when the CVDF mission is completed.

Safety Considerations

The main structures of the CVDF provide the following functions: secondary confinement in conjunction with the HVAC exhaust systems (except the administrative building), radiation protection (as low as reasonably achievable [ALARA] exposure from cask-MCO and contaminated effluents), fire protection, natural phenomena and hazard mitigation (seismic, tornado and high winds, floods), and worker safety hazards protection (industrial safety and industrial hygiene).

B2.5 PROCESS DESCRIPTION

This section contains information on the CVDF processes and systems. Section B2.5.1 provides a summary-level integrated process description. The remaining sections contain CVDF system process descriptions. The intent of this information is to provide an understanding of normal operations of the facility.
The purpose of cold vacuum drying is to remove free water from the MCO. Free water and certain loosely bound waters of hydration will be removed by cold vacuum drying. More strongly bound waters of hydration and chemically bonded forms of water will not be removed by the cold vacuum drying process. A generalized drawing of the types of water contained within an MCO and the processes to be used for removal of these waters is provided in Figure B2-9. The chemically bound water that remains following cold vacuum drying continues to decompose via radiolysis, causing pressure to increase in the sealed MCO during storage (HNF-1523). In HNF-SD-SNF-TI-040, MCO Internal Gas Composition and Pressure During Interim Storage, it was determined that the amount of bound water remaining after completion of cold vacuum drying will not be sufficient to result in overpressurization of an MCO during storage at the CSB. This FSAR addresses the safety of the CVDF drying process and does not address the potential pressurization of the MCO after the MCO leaves the CVDF for transfer to the CSB.

B2.5.1 Cold Vacuum Drying Process Overview

The following process overview summarizes process functions performed from MCO receipt through final testing and shipping. CVDF system and equipment operating procedures guide all processing activities. CVDF personnel receive facility- and procedure-specific training. Figure B2-10 provides a block diagram of the cold vacuum drying process. Figure B2-11 shows the progression of water removal from an MCO during the cold vacuum drying process. The simplified diagram of cold vacuum drying process systems (Figure B2-12) serves as a preliminary overview of the process systems that are utilized during MCO cold vacuum drying. HNF-1851, Cold Vacuum Drying Residual Free Water Test Description, provides more detailed information on the process parameters, and their bases, associated with the cold vacuum drying process discussed in the remainder of this section.

B2.5.1.1 Multi-Canister Overpack Transport Receipt

Prior to cask-MCO receipt via tractor-trailer, CVDF operators ensure required systems and equipment are operational and facility technical safety requirements are met. For receipt, the SCIC system is in (1) BYPASS. The shipping package is verified for quality assurance acceptability. Upon receipt, the trailer is positioned in the process bay and the cask-MCO trailer is disconnected. The tractor is driven out, the telescoping door is closed, and the trailer is leveled. Radiation surveys are performed and exposure rates are posted. Security systems are established. The mezzanine bridge and work platform hand rails are configured for personnel safety.

B2.5.1.2 Cask Venting and Lid Removal

The cask-MCO is shipped with the MCO filtered process exit port valve in the open position, which allows the cask headspace to pressurize and collect hydrogen during shipping. Prior to cask lid removal, process vent lines are evacuated, the cask headspace is sampled (primarily for hydrogen concentration) and the cask undergoes helium pressurization and venting. Hydrogen is sampled during process validation using a sample cartridge that is sent to a lab for analysis and trending. The sample results are used for design confirmation and are not required in order to continue with the cold vacuum drying process.
The cask lid is removed using an overhead process bay bridge crane. The process vent plug valve on the MCO is then closed manually.

**B2 5 1 3 Process Hood-Seal Ring Installation** Installation of process connections are made after moving the process hood into place and removing the process port covers from the MCO. As received from the K Basins, the long axial process tube port valves is in the closed position while the filtered process exit port is received in the open position but is closed prior to installation of the hood/seal ring. The process hood-seal ring is lowered onto the top of the MCO using the process bay crane. When in position, the ring is bolted to the top of the cask. Following installation, two air-inflated seals provide annulus water confinement between the cask and the MCO. The process hood open-faced ventilation inlet is incorporated into the process hood structure. A flex hose connects the open-faced ventilation inlet with the process bay local exhaust HVAC and process vent system. The process hood also provides staging locations for the MCO VPS process connectors. A portion of the seal ring (leak test fixture) also encloses the main MCO seal ring area. Vacuum is applied during the cold vacuum drying process to remove residual water on the upper seal ring to facilitate helium leak testing at the conclusion of cold vacuum drying processing. Figure B2-13 shows the system interfaces with the MCO, and Figure B2-14 shows a detailed view of the process hood and seal ring installation on the MCO.

**B2 5 1 4 Vacuum Purge System Hookup and Venting** The process connectors are designed to provide connections to both the VPS and the helium systems through flex piping. Following installation, the process connectors are pressurized with helium and a leak test is performed. When leak-tight conditions are verified, the VPS system is purged with helium and vented to reduce residual air in the suction lines. The MCO process vent plug valve is then opened and helium pressurization and venting cycles are performed to reduce the hydrogen concentration in the MCO headspace. Pressurization and venting of the MCO headspace occurs prior to starting the drain cycle.

**B2 5 1 5 Multi-Canister Overpack Tempered Water System Hookup, Heatup and Drain, and Purge/Flush** For MCO heatup and drain, the SCIC system position switch is moved to (2) HEATUP. The gas-operated valves for the PWC line are interlocked closed to prevent premature draining of the MCO. Tempered water (annulus) system connections are established with the cask through the lower cask port and upper tempered water seal ring. Recirculating tempered water flows through the cask annulus from the bottom to the top at a flow rate of 15 to 20 gal/min and pressure of 20 to 35 lb/in² gauge. Prior to commencement of the drain a period of time is allowed for the MCO to “soak” so that the MCO contents can warm up and the water in the tempered water annulus can reach its operating range between 40 °C and 50 °C. Following the “soak,” the MCS is instructed to begin the draining sequence. The process water receiver tanks are purged (also referred to as a PWC prepurge) with helium to remove oxygen in the tanks. The SCIC system is activated by key switch to enter the (3) DRAIN mode. This action allows the MCS to begin the drain sequence. The MCO drains through the long process tube port to the PWC system. The drained water is treated by the PWC system, stored in a holding tank, and ultimately returned to the K Basins. While the water is draining, the helium supply system pressurizes the MCO through the MCO filtered process exit port. Draining takes...
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approximately 30 minutes to remove the water from the MCO. Upon detection of water suction breakthrough (large increase of PWC system inlet pressure) the drain lineup remains unchanged for several minutes to remove residual water. The SCIC system position switch is moved to (4) PURGE/FLUSH. This automatically initiates a SCHe line purge of the long process tube to clear collected water in the line. After the SCHe line purge, the MCS starts a PWC system line flush with deionized water. The MCO pressure is normally maintained via helium through the filtered process exit port. The deionized water rinse runs deionized water to the receiver tanks. After the deionized water flush, helium is used to perform a PWC postpurge of the PWC system tanks to remove hydrogen that may have collected during the drain sequence. The process helium supply to the MCO is valved to flow from the MCO long process tube port, through the MCO, and out the filtered process exit port to the process vent system. This is the normal flow path for all drying mode operations.

B2 5 1 6 Vacuum Purge Drying Operation After draining and purge/flush, the SCIC system key switch is moved to the (5) DRYING mode. The gas-operated valves for the deionized water line are interlocked closed by the SCIC system to ensure against inadvertent water addition. During normal operations, the MCS automatically controls the drying operation. Verification is made that the MCO is being pressurized with process helium supplied via the MCO long process tube port and out the HEPA filter and filtered process exit port prior to the vacuum purge drying operation. The initial vacuum drying sequence dries the MCO contents using the VPS vacuum pump and condenser. The vacuum pump is used to remove noncondensable gases from the MCO cavity while a condenser is used to condense water vapors. Drying operations alternate positive pressure and vacuum drying modes with a limit of less than 8 hours for the first vacuum cycle and then 4 hours for all successive pressure and vacuum cycles. If during the drying process the 8-hour vacuum timer limit is reached, the SCIC repressurizes the MCO with a 4-hour minimum helium pressurization and purge. During vacuum pumping at pressures greater than a specified value, a small helium purge is normally added for hydrogen dilution. When the condenser differential temperature has reached a minimal setting for several minutes, the MCS isolates the condenser via the condenser isolation valves because the amount of water being removed from the MCO is relatively small. At this time, the process may proceed to the no-purge vacuum check.

B2 5 1 7 No-Purge Vacuum Check A no-purge vacuum check is performed after VPS condenser delta temperature reaches a minimal setting. With the VPS condenser valved out and the helium purge isolated, MCO pressure must decrease to less than 8 torr within 5 minutes. If 8 torr is not reached in 5 minutes, then the MCO is returned to the normal drying sequence, and the no-purge vacuum check is performed every hour while under vacuum. If the pressure decreases to less than 0.5 torr for a minimum of 15 minutes, the pressure rebound pretest is conducted. If not, the system is returned to the vacuum purge drying cycle.

B2 5 1 8 Pressure Rebound Pretest If the no-purge vacuum check is passed, a pressure rebound pretest is performed with the vacuum pump isolated. If the MCO pressure rises less than 0.4 torr over a 10-minute period, the test continues. If not, the system is returned to the vacuum purge drying cycle.
B2 5 1 9 Initial Pressure Rebound Test If the pressure rebound pretest is passed, a 1-hour pressure rebound test is conducted to indicate whether free water remains within the MCO. The pressure rebound test is performed after the MCO has undergone a 4-hour helium repressurization (thermal reset). This test is run from the vacuum system base pressure (expected to be less than 0.5 torr) with the MCO isolated. The tempered water inlet temperature must be maintained between 40 °C and 50 °C. Vacuum pressure and tempered water temperatures (inlet and outlet) are recorded. If the pressure increases at an average rate greater than 2.4 torr/h, the test fails and more vacuum purge drying is necessary.

B2 5 1 10 Proof of Dryness Demonstration Following a successful pressure rebound test, the SCIC system switch is placed in the (6) PROOF mode. The gas-operated valves for the VPS line continue to be interlocked closed to ensure against inadvertent water addition. The tempered water temperature is maintained between 40 °C and 50 °C. The goal of this step is to maintain the MCO under vacuum less than 18 torr for a minimum cumulative period of 8, 20, or 28 hours for no scrap basket, one scrap basket, or two scrap baskets respectively. Should the pressure increase above 18 torr, the MCS will automatically start a helium purge. An increase in pressure indicates an offgasing of hydrates, a leak of air or helium, additional free water in the MCO being released, or potential offgasing from other sources. The residual gas monitoring system (see Section B2 5 3 3) may be used to determine the source of pressure. The system can be returned to the vacuum purge drying cycle, or recovery procedures can be initiated if proof of dryness is not shown.

B2 5 1 11 Final Pressure Rebound Test The final pressure rebound test is used to confirm that bulk water was not reintroduced during the vacuum drying cycle. When the proof-of-dryness demonstration has been completed, the MCO is isolated from potential water sources (the vacuum pump and condenser systems), the VPS line gas-operated valves continue to be interlocked from the previous (6) PROOF mode position, and the pressure rebound test is repeated a final time. The test is successful if the pressure increases at an average rate that does not exceed 2.4 torr/h (HNF-1851). Following a successful test, drying is complete and the MCO is pressurized with helium. These data are important for long-term storage of SNF and are designated as Office of Civilian Radioactive Waste Management data.

B2 5 1 12 Multi-Canister Overpack cooldown The MCO requires cooling to approximately room temperature before final preparation for shipping. Cooldown is achieved by starting cooling water flow through the tempered water system cooler with the heaters off. After a mandatory hold is completed, the MCO helium pressure is verified and the MCO process port valves are closed sequentially.

B2 5 1 13 Multi-Canister Overpack Helium Leak Test and Preparation of Cask for Departure MCO process connectors are disconnected. A mass spectrometer leak detector (MSLD) leak check is performed on the MCO process ports and MCO main seal. The annular space between the MCO and the cask is drained to the PWC system and then the cask–MCO annular space is isolated from the tempered water system. The cask–MCO annulus is purged with instrument air to dry. The MCO helium MSLD and auxiliary vacuum system leak test are performed to verify that the MCO leak rate is <1 x 10^-5 standard cm^3/sec. The process hood-seal...
ring is removed, the cask and MCO exposed top surfaces are dried and cleaned, and radiation and contamination surveys are performed. The cask lid is installed. The trailer is prepared for transport, standard pretrip inspections conducted, and the cask, along with the MCO quality assurance package, is released for transfer to the CSB.

**B2 5 2 Cask-Multi-Canister Overpack Systems**

**Summary Description**

The transportation cask, trailer, and MCO are not part of the CVDF but are described in the following subsections because of the integral function they provide to the overall CVDF processing activities. The physical interfaces between the CVDF and the cask and MCO are as follows:

- Process bay local exhaust HVAC and process vent system (Section B2 6 2 4)
- PWC system (Section B2 5 7 1)
- Vacuum and purge system (Section B2 5 3)
- Deionized water system (Section B2 9 5)
- General-service helium system (Section B2 5 4 1)
- SCHe system (Section B2 5 4 2)
- Tempered water (annulus) system (Section B2 5 8 1)
- CVDF structure (process bay) (Section B2 4)
- Instrument air (Section B2 9 2)

A simplified diagram showing the CVDF system interfaces with an MCO is presented in Figure B2-13.

**B2 5 2 1 Transportation Cask and Trailer**

A design description of the transportation cask and trailer is provided in HNF-SD-TP-SARF-017. Additional details describing the interface between the MCO, transportation cask, trailer, and the CVDF are provided in HNF-SD-SNF-DRD-002.

SNF is received at the CVDF packaged in an MCO that is transferred from either the K East or K West Basin in a cask loaded on a tractor-trailer. The MCO, containing fuel and flooded with water, remains in the cask and on the trailer throughout processing in the CVDF. The trailer is disconnected from the tractor before processing activities are started. The trailer design includes platforms that serve as work stations for CVDF operators and provides direct access to cask-MCO process interface connections.

**B2 5 2 1 1 Transportation Cask**

The transportation cask (i.e., cask) is a category 1 package (NRC Regulatory Guide 7 11, Table 1). Therefore, NUREG/CR-3854, *Fabrication Criteria for Shipping Containers*, indicates that the design, fabrication, and testing of the cask containment boundary are performed to the intent of the requirements of the *Boiler and Pressure
Vessel Code, Section III, Subsection NB, Class 1 (ASME 1995) The containment boundary design follows the criteria from Section III of the code The structural analyses criteria meet Section III requirements However, the cask is not code stamped

Fabrication of the cask is in accordance with the guidelines of NUREG/CR-3854 All welds and weld joints meet NUREG/CR-3019, Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials criteria and were examined in accordance with the Boiler and Pressure Vessel Code, Section III, Subsection NB (ASME 1995) All containment welds are radiographed per the Boiler and Pressure Vessel Code, Section III, Subsection NB-5000

Inspections and containment leak testing of the cask were performed per Boiler and Pressure Vessel Code, Section III (ASME 1995), and ANSI N14.5, Radioactive Materials — Leakage Tests on Packages for Shipment during fabrication

The cask is a vertical, cylindrical, stainless steel vessel The overall packaging assembly, including the lifting device, has an outside diameter of approximately 44 in and an overall height of 190 in The cask consists of a forged, stainless steel cylinder with an integrally welded stainless steel bottom head that is 6-in thick

The cask lid is forged stainless steel The lid is bolted to the cask body with 12 bolts that are arranged on a circle A single, butyl rubber O-ring seal forms the confinement boundary between the cask body and lid Lid installation is guided by two alignment pins that are integral to the cask body Two lifting brackets are welded to the cask lid for lifting of the cask and the lid A 4-in diameter trunnion is welded to each lifting bracket

The cask also has two vent ports and one drain port The vent ports are on the exterior of the cask lid, while the drain port is positioned in the side of the cask body The drain port is closed with quick-disconnect couplings The drain port has a cover that is bolted onto the cask with four bolts One vent port is closed with a quick-disconnect coupling that has a cover that is bolted onto the lid with four bolts The other vent port is closed with a head cap screw and has a cover that is threaded into the lid with a "snout"

B25212 Trailer The trailer serves as a support mechanism for the cask—MCO during the vacuum drying process It is a three-axle trailer and is attached to a standard tractor The trailer was procured as a commercial-grade item

The tiedown system on the trailer is a fixed system, which is an integral part of the conveyance system The conveyance system is based on a custom double-drop semi-trailer designed specifically to secure and transfer the cask The cask is transferred in the vertical position and fits into the cask support device, an approximately 15-in-deep well located beneath the deck of the trailer Approximately 118 in above the deck of the trailer, the cask is secured by a cask tiedown device mounted onto a fixed-frame constructed of structural tube members that are braced to the trailer deck with four structural tube members The tiedown system is
constructed of a rectangular, structural steel tubing (ASTM A500-96, Grade B) frame that is welded to the transport trailer bed. The cask tie-down device is designed as a hinged clam-shell ring where each half section pivots about a fixed hinge pin and is secured with three hex head bolts. The inside diameter of the clamping ring is constructed of 6061-T6 aluminum and equipped with a neoprene abrasion pad that forms a tight fit with the cask. Attached to the clamping ring are four cask hold-down brackets, equally spaced around the circumference, that restrain vertical movement of the cask.

The overall length of the trailer is 40 ft, its overall width is 10 ft, and its overall height with cask in place is 17 ft, 6 75 in. The trailer has a work platform that is permanently attached to it. The work platform can be used to work around the top of the cask and MCO while at the CVDF. The trailer also provides access to the bottom drain port.

**B2 5 2 2 Multi-Canister Overpack** Full MCO design details and design requirements, as well as detailed discussions of fuel inventories that are to be contained within the MCOs, are presented in HNF-SD-SNF-SARR-005. Additional information on the design of the MCO is provided in HNF-SD-TP-SARP-017. The information summarized below is to describe for the reader the basic design of an MCO and the anticipated fuel inventory contained therein.

The MCO is a single-use container that consists of a cylindrical shell, five or six fuel baskets, a shield plug, and design features necessary for maintaining the structural integrity of the container while it provides SNF criticality control and fuel drying capability. HNF-S-0426, *Specification for Spent Nuclear Fuel, Multi-Canister Overpack*, implements the functional requirements documented in WHC-SD-SNF-FRD-016, *Spent Nuclear Fuel Multi-Canister Overpack Technical Functions and Requirements*, which defines the MCO, and thus establishes the minimum essential requirements that the MCO design must meet. A diagram of a typical MCO assembly (with five fuel baskets) is provided in Figure B2-15.

The MCO provides confinement of SNF and maintains the SNF in a critically safe configuration. NRC requirements (NUREG-1536) for a Title 10, *Code of Federal Regulations*, Part 72 "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR 72) licensed facility would allow for application of either Subsection NB or NC, 3 IV 1 b(1)(b) of the Boiler and Pressure Vessel Code (ASME 1995). The MCO's structural design has been developed and the MCOs have been manufactured and code stamped in accordance with the Boiler and Pressure Vessel Code, Section III Subsection NB.

Key parameters required for maintaining confinement include the differential temperatures established in the MCO during processing. The maximum temperature difference between the outside of the MCO shell and the center of the shield plug must be less than or equal to 100 °C (180 °F) and less than or equal to 5 °C (9 °F) radially within the MCO shell wall. These parameters limit thermal expansion and preserve the integrity of the MCO mechanical seal.
Thermal analysis has shown that these key parameters are never approached during any of the MCO processing steps (HNF-SD-SN-SARR-005). The normal temperature range for the non-pressure her annulus water ensures the differential temperatures are well within the limiting values. Consequently, there is no requirement for operational controls to ensure that the differential temperature parameters are not exceeded.

Several types of fuel elements are contained in an MCO that will be processed through the CVDF: Mark IA, Mark IB, Mark IC, Mark IV, and Mark IVB. These types of fuel elements are contained within baskets, which are in turn contained within an MCO. The MCO fuel baskets are categorized into two major types: intact fuel element baskets and scrap fuel baskets. The Mark IA fuel and scrap baskets are designed in accordance with the Boiler and Pressure Vessel Code, Section III, Subsection NG (ASME 1995), under the component safety group as guided by NUREG/CR-3854. Mark IA fuel has a higher $^{235}$U enrichment than Mark IV fuel. Structural integrity is required of the Mark IA basket center post for criticality control whereas structural integrity for criticality control is not required for the Mark IV fuel basket. Therefore, the more stringent Section III, Subsection NG requirements are applied to the construction of the Mark IA fuel and scrap baskets. The remaining fuel elements will be handled in accordance with the requirements of Chapter C6.0 of Annex C, the K Basins FSAR.

There are four different basket types that may be contained within an MCO:

- **Type 1**—Holds 48 intact Mark IA fuel elements
- **Type 2**—Holds 54 intact Mark IV fuel elements and does not need an exclusion void
- **Type 3**—Holds Mark IA fuel fragments
- **Type 4**—Holds Mark IV fuel fragments and does not need the exclusion void

Type 1 and Type 3 baskets contain a center post to provide a critically favorable geometry in the baskets and the MCO.

The MCO has a shield design that protects workers from gamma rays and neutrons emanating from the inside of the MCO. The shield plug provides access to the interior of the MCO via four penetrations (Figure B2-16). The penetrations accommodate two process ports (ports 2 and 3), a safety-class rupture disk port (port 4), and a spare port (port 1). The two process ports have valve mechanisms to accommodate connections to external equipment. These process ports connect internally to two process paths. One process path (port 3) connects to a long process tube that extends down the MCO axis to the bottom and the second process path (port 2) extends to internal filters mounted on the shield plug's underside. A third path (port 4) extends from the rupture disk to the space between the bottom of the shield plug and the top of the guard plate. The long process pipe and the guard plate are fitted with debris screens. The connections leading to the long process tube or the filtered process exit port are designed to be easily differentiated by a worker looking at either the top or bottom of the shield plug.
The shield plug features an integrally machined axisymmetric lifting ring with a 12-ton lifting capacity when gripped with six equally spaced grippers. The ring facilitates handling of the MCO at the CSB.

The MCO has an internal filter to support the vacuum drying outflows from the MCO. The filter is installed between the shield plug bottom and the guard plate. The internal filter prevents fuel particulate corrosion products from leaving the MCO (thereby reducing dose in the facility) and reduces the risk of external contamination during breaking of CVDF process piping connections. The internal filter is a HEPA filter and adequate flow capacity is achieved through the filter to support operating needs. The filter does not meet the testing provisions of DOE Order 6430 1A because its efficiency cannot be tested after the shield plug is installed. The filter is not relied upon to perform a safety function. No credit has been taken for the presence or function of this filter in the accident analysis.

**Safety Considerations**

The cask functions to provide a flooded shipping container for MCO transport from the K Basins to the CVDF and to provide radiation shielding, a safety-significant function for worker safety. While at the CVDF, the cask provides the safety-class function of maintaining a water-filled annulus for heat transfer purposes.

The MCO provides criticality control functions for the Mark IA fuel. Within a closed cask, the MCO is designed to withstand a 30-min, 802 °C (1,475 °F) fire and still meet the radioactive material release criteria of Title 10, Code of Federal Regulations, Part 71, “Packaging and Transportation of Radioactive Material” (10 CFR 71), Section 71.51(a)(2) (HNF-SD-TP-SARP-017).

B2.5.3 Vacuum and Purge System

The cold vacuum drying process drying mode uses the VPS to place the MCO contents under vacuum and to perform inert helium gas purging using a cycle of helium flow, evacuation, and inert gas backfill/pressure stages. While in process, the VPS operation includes monitoring and testing stages to demonstrate acceptable equipment and MCO conditions. The acceptable conditions must be met before the process is allowed to begin or continue. The vacuum and purge system consists of several subsystems that function together. See Figure B2-17 for the process systems flow diagram.

**Summary Description**

The VPS process equipment is designed in a modular fashion consisting of one process hood and piping assembly and one process equipment skid.
The process equipment skid is an assembly of vacuum pump, water pump, residual gas monitoring (RGM) system, tanks, heaters, pipes, valves, instruments, instrumentation wiring, and electrical distribution wiring mounted on a frame that rests on the floor of a process bay. A mezzanine is located above the process equipment skid at the appropriate height to access the top of the MCO. A process hood and piping assembly serves as the key interface between the MCO and the process equipment skid. The process hood consists of a stainless steel slot hood designed to be bolted to a seal ring subassembly and, in turn, to the MCO and cask. See Figure B2-14 for a diagram of the process hood and seal ring installation on the MCO. Included with the hood and piping assembly are process connectors (containing valve operators) designed to be bolted directly to the MCO and to serve as a piping connection and tool for opening and closing process port valves in the shield plug. The process hood provides protection to operators by capturing radioactive particulate matter that may be released from the MCO while attachments of the process connectors are made or broken. The process hood also provides a location for the temporary storage of the process connectors while not in use. The seal ring subassembly is designed to be bolted to the top of the MCO and shield cask and to seal the interstitial space between the MCO and cask. The seal ring subassembly also has connections to facilitate manual refill of the cask-MCO annulus. Process piping associated with the assembly is mounted on a stand attached to the mezzanine and includes pipe, valves, and instrumentation.

VPS operating limits and normal operating conditions are provided in Table B2-3.

<table>
<thead>
<tr>
<th>Flow (through filtered process exit port)</th>
<th>Operating limits</th>
<th>Operating conditions</th>
</tr>
</thead>
<tbody>
<tr>
<td>No minimum</td>
<td>Variable</td>
<td></td>
</tr>
<tr>
<td>100 ft³/min maximum</td>
<td>100 ft³/min</td>
<td></td>
</tr>
<tr>
<td>Temperature</td>
<td>10 °C minimum</td>
<td></td>
</tr>
<tr>
<td></td>
<td>50 °C maximum</td>
<td></td>
</tr>
<tr>
<td>Pressure</td>
<td>0.05 torr minimum</td>
<td></td>
</tr>
<tr>
<td></td>
<td>150 lb/in² gauge maximum</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Variable</td>
<td></td>
</tr>
<tr>
<td></td>
<td>10 lb/in² gauge ± 15%</td>
<td></td>
</tr>
<tr>
<td></td>
<td>(at filtered process exit port)</td>
<td></td>
</tr>
</tbody>
</table>

The use of stainless steel VPS piping and valves ensures that there will be negligible corrosion over the design life of the CVDF. A corrosion allowance of 0.06 in is specified in the ASME code data reports for the VPS equipment.

The process equipment skid occupies a space 18 ft east-west, 6 ft north-south, and 10 ft high. The skids are located 3 ft from the north wall of each process bay, centered between the east and west walls. The process hood support stand is 10 ft east-west, 3.5 ft north-south, and...
7 ft high, centered east-west and is attached to a base frame located near but isolated from the north mezzanine overhanging each process bay.

Figure B2-18 provides an overhead view of the location of a cask-MCO trailer in a process bay in relation to the process skid, process hood/seal ring and support, and mezzanine. Figure B2-19 shows a side view of the process skid (skid mezzanine, and process hood and support) in a typical process bay.

The vacuum and purge system consists of the following subsystems:

- **VPS** — The VPS consists of a single four-stage vacuum pump piping, valves, and pressure control valves and piping for helium purge gas supply and venting.
- **Heat trace system** — The heat trace system is used to minimize condensation in piping ahead of the VPS condenser and collection tank.
- **RGM system** — The RGM system consists of instrumentation and gas monitoring equipment that monitors the operating pressures and gas concentrations.
- **Helium MSLD and auxiliary vacuum system** — The helium MSLD and auxiliary vacuum system measures the pressure boundary leak rate of the MCO following the cold vacuum drying process.

The tempered water (annulus) system is on the process equipment skid but is treated as a separate process system in this document (see Section B2 5 8 1).

**B2 5 3 1 Vacuum Purge System**

The VPS has 11 major interfaces with the other CVDF systems. The interface systems, in addition to the cask-MCO, are as follows:

1. Electrical power distribution system (Section B2 8 1)
2. SCIC system (Section B2 5 9 2)
3. Deionized water system (Section B2 9 5)
4. Instrument air system (Section B2 9 2)
5. Facility structure (skid and process hood support stand) (Section B2 4 1)
6. Chilled water system (Section B2 5 8)
7. General-service helium system (Section B2 5 4 1)
8. SCHe system (Section B2 5 4 2)
9. Process bay local exhaust HVAC and process vent system (Section B2 6 2 4)
10. PWC system (Section B2 5 7 1)
11. MCS (Section B2 5 9 1)

The MCO is connected to VPS hardware via flexible hoses that run from the MCO process connectors to valves on the process hood support stand. Helium is supplied to the MCO interior through two separate paths (depending on the process mode). The MCO venting flow is through...
the MCO vent port, which is provided with HEPA filters inside the MCO shield plug. The flow then travels to the condenser and condenser tank, or is bypassed around the condenser under some vacuum conditions. The flow then proceeds through the vacuum pump to the process vent system that discharges to the HEPA-filtered process bay local exhaust HVAC and process vent system.

A 1-in VPS line is connected to each of two of the MCO ports. One of the lines (long process tube port) connects to the general-service helium system and deionized water line and also has connections to the SCHe system. The other line (filtered process exit port) provides the pathway directly to the VPS equipment with appropriate connections to the SCHe system. In general, the VPS consists of an RGM, a condenser (which can be bypassed after most of the water is removed from the MCO), a condenser water collection tank routed to the PWC system, a helium line to provide pressurization to remove MCO water, and a four-stage roots vacuum pump that connects to the process vent. When water has been drained from the MCO, a valve to the vacuum pump is opened and pump-down is initiated to remove the remaining water in the MCO fuel. At various stages the water/gas content of the MCO is monitored with the RGM and the condenser captures water vapor before it reaches the vacuum pump.

The VPS is designated general service but certain components are designated safety class. The safety-class components discussed in Chapter B4 0 are designed to function under worst-case internal and external environmental conditions. The VPS safety-class components that are credited with performing their safety function in a seismic event are seismically qualified to performance category 3 criteria. The VPS is protected from high winds by the CVDF structure. All VPS safety-class isolation valves have remotely activated operators and fail closed upon loss of control air, control signal, or electric power.

General service VPS tanks, valves, components, instrumentation, and controls are designed and qualified for performance category 1 as defined in DOE-STD-1020-94. Structural components and equipment anchorage are designed to be performance category 2 on the process equipment skid and performance category 3 on the process hood support stand.

The VPS has sufficient designed testability features to permit the periodic measurement and calibration of all setpoints and adjustments that affect the manner in which the VPS performs. Periodic testing of VPS SSCs is dictated by the requirements of the individual components according to the respective manufacturer's recommended schedule and practice or industry standards and is administered by controlled procedures.

The majority of VPS operation is automatically controlled by the MCS, but manual operations in the process sequences include field operator actions (e.g., connecting the MCO valves operating manual block valves, connecting quick-disconnect valves) and control room operator actions (e.g., acknowledging alarms, operating the SCIC system mode key switch). Operators direct the MCS to initiate a preprogrammed sequence in the MCS once conditions have been satisfied. Valve state changes, gas supply pressure control, process status notifications, and system alarm notifications are performed by the MCS or the SCIC system.
**B2.5.3.2 Heat Trace System** The heat trace system is installed on the VPS piping in each of the four process equipment skids and on the process hoods to minimize condensation and icing in piping ahead of the VPS condenser and collection tank. The heat trace system consists of (1) self-regulating heat cable made up of two bus wires, a semi-conductive polymer core of which resistance varies with temperature, and an insulating jacket, (2) connection accessories, and (3) local on-off thermostatic control. The heat trace cable has a fluoropolymer jacket over braid and is controlled from a local controller with resistance temperature detector input.

**B2.5.3.3 Residual Gas Monitoring System** The RGM system, during drying, provides continuous online monitoring of MCO offgas composition to assist in evaluation of fuel drying effectiveness and sends data to the MCS, which initiates an alarm if there are gas concentrations outside of process limits. The CVDF contains four RGM subsystems, one in each process bay. Most of the RGM subsystem is enclosed in a case and mounted on the VPS process equipment skid. RGM subsystem components consist of a pump, ionizer, filter, collector, and quadrupole-type mass spectrometer with associated piping, valves, instruments, alarms, and controllers.

The residual gas sample is drawn from the MCO vent line and analyzed for hydrogen, nitrogen, oxygen, argon, helium, water vapor, krypton, and xenon. Sample gases are discharged from the RGM system by a vacuum pump through a vent line to the CVDF process bay local exhaust HVAC and process vent system.

Normal RGM system operation interfaces with the MCS. Results of this monitoring are used along with pressure and temperature measurements of MCO gases to evaluate (1) adequacy of the vacuum conditioning process in removing water from the MCO fuel, (2) any air inleakage to the system, (3) the gas constituents of the MCO, and (4) presence of radioactive gas species. The system is designated as performance category 1.

**B2.5.3.4 Helium Mass Spectrometer Leak Detection System** The helium MSLD and auxiliary vacuum system are used to measure MCO leak tightness at the conclusion of the cold vacuum drying process to verify that the MCO is ready for interim storage. This system consists of a helium MSLD system, auxiliary vacuum pump, piping, valves, and instrumentation to measure the pressure boundary leak rate of the MCO following the cold vacuum drying process.

**Safety Considerations**

The VPS piping and valves perform confinement isolation functions between the MCO process connectors and isolation valves located on the process hood support stand, are classified as safety class and are designed to performance category 3 standards. Because the VPS is directly connected to the MCO, there is a potential for radioactive contamination throughout the system piping and components.

All of the VPS safety-class components required for process instrumentation and required to accomplish MCO isolation during process upset conditions are included in the process hood support stand. The safety-class components consist of the following:

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*November 1999*
Fail close MCO isolation valves with instrument air line filters mounted on the process hood support stand

- Pressure transmitters, pressure indicators, and flow indicating transmitters mounted on the process hood support stand

- MCO 30 lb/in² gauge rupture disk and associated vent path to a local exhaust duct

- Piping inside isolation valves

A safety-class vent path mitigates the effects of a pressurized release from the MCO due to a postulated overpressurization event. The vent path originates from the MCO process ports number 2 (filtered process exit port) and number 4 (150 lb/in² gauge rupture disk). The primary vent path is accomplished via the 30 lb/in² gauge rupture disk on the MCO port number 2 process connector. A check valve with a 30 lb/in² gauge cracking/reseal pressure setting, is located downstream of the rupture disk. The primary vent path discharges to the process bay local exhaust HVAC system via a tie-in downstream of the process hood's bubble-tight isolation damper. A secondary vent path is provided for the MCO through port number 4 and its associated 150 lb/in² gauge rupture disk. This secondary vent path is not directly tied into the process bay local exhaust HVAC system. A diagram of the safety-class vent configuration for the cask-MCO is provided in Figure B2-13.

The deionized water flush line has gas-operated isolation valves with interlocks to isolate the MCO from the deionized water source for times other than when the MCO is undergoing preparations for cold vacuum drying, MCO draining, and PWC system line flushing. The PWC has interlocks to prevent draining during the (2) HEATUP mode.

The seal ring subassembly has safety-class connections to facilitate manual refill of the cask-MCO annulus, should a loss of annulus water occur.

B2 5.4 Helium Systems

Two helium systems are provided in the CVDF. The helium supply is normally from a commercial tube trailer. A backup supply is provided from the SCHe system gas cylinders. Each helium system is described in the following subsections.

B2 5.4.1 General-Service Helium System

Summary Description

The function of the general-service helium system is to provide pressurized helium for process operations, process support activities, and process safety functions (when available). Helium is an inert gas with good thermal conductivity and is used in process operations to provide
enhanced thermal conduction, to purge the MCO of hydrogen and other gasses, to pressurize the MCO to preclude oxygen ingress, and to provide an inert backfill of the MCO when drying is complete. The general-service helium system is used to provide thermal conductivity within the MCO and to enhance fuel corrosion reaction heat removal, to inert process vessels, to purge various process lines, to pressurize the MCO headspace during draining, and to provide an inert gas to pressurize the labyrinth seals on vacuum pumps.

The general-service helium system also provides (1) a non-safety-class helium supply to the SCHe system (see Section B2.5.4.2) (2) helium flow information to the SCIC system (see Section B2.5.9.2), and (3) safety-class isolation of the MCO. The general-service helium system is designed to supply helium in sufficient quantities and pressure to support simultaneous process operations in all process bays.

The general-service helium system is located both inside and outside the CVDF. Two leased helium tube trailers are parked on a concrete pad located on the west side of the CVDF. Only one helium tube trailer is used to supply helium to the facility at any one time. A helium pipe penetrates the west wall and connects to a helium header running north-south along the full length of the east wall of the transfer corridor. Figure B2-20 shows the general-service helium system delivery to the CVDF process bays. Branch piping connects each process bay to the helium header and flexible piping is used to distribute the helium to the process equipment skid, the process hood and support stand, the VPS, and the SCHe system. Within the process hood and support stand, the general-service helium system is connected to the VPS lines connected to the long process tube and the filtered process exit port.

In addition to piping, the general-service helium system includes pressure regulators, pressure and flow indicators, and control valves. System control and alarm functions are provided by the MCS (see Section B2.5.9.1). Check valves are used to prevent backflow from one process bay reaching the other process bays. The two tube transporters provide, at a minimum, a two-week supply of helium at the maximum process demand rate. Continuous operation is provided by allowing an empty tube transporter to be disconnected and replaced while helium is drawn from the other.

The general-service helium system has redundant safety-class safety relief valves designed to relieve pressure in the event of pressure regulator failure, as well as a general service safety relief valve.

Safety Considerations

The general-service helium system has safety-class components that provide for isolation of the MCO (e.g., redundant fail-closed isolation valves with associated instrument air filters and flexible tubing connections) and safety-class safety relief valves. The general-service helium system also has redundant safety-class instrumentation for process operation, identification of process upset conditions and initiation of SCIC system functions.
B2 5 4 2 Safety-Class Helium System

Summary Description

The SCHe system is a dedicated safety system that actuates upon SCIC signal trip, which can be automatic or manual, or upon loss of power or air. The key function of the SCHe system is to pressurize and purge the MCO with sufficient helium pressure and flow to preclude flammable concentrations of hydrogen and oxygen.

The SCHe system is capable of performing the following functions: (1) remove hydrogen and/or oxygen to maintain the MCO at less than the lower flammability limit, (2) maintain positive pressure in the MCO to preclude or limit oxygen ingress, (3) pressurize the MCO to provide enhanced thermal conductivity within the MCO, and (4) provide a pressure regulated vent flow path.

There are independent and identical SCHe systems for each process bay, the following description is applicable to all systems. Each process bay SCHe system consists of four parallel purge systems: two on the VPS line going to the MCO long process tube and two on the VPS line going to the MCO filtered process exit port. Each purge system consists of helium supply bottles, pressure regulating valves, rupture disks, and fail-open electropneumatically operated isolation valves with associated piping and instrumentation. The four lines form two redundant purge, vent, and pressurization paths for the MCO.

Each of the four SCHe system trains in a process bay has two sources of helium, a 240 ft³ (at standard temperature and pressure) helium bottle, and the normal helium tube trailers associated with the general-service helium system. The safety-class bottles provide the helium supply should the non-safety-class process tube trailer be unavailable. Figure B2-19 shows a side view of a process bay with the location of SCHe bottles in relation to the process skid. Figure B2-21 shows an overhead view of the location of the SCHe bottles in relation to the process skid within a process bay.

Pressure control settings are used to preferentially use general-service helium from the tube trailers. Under nonfaulted conditions, this preserves the SCHe system purge bottle supply and reduces the need for SCHe system bottle refill. Double check valves and a pressure regulator on the general-service helium supply to the SCHe are set slightly higher than the regulator for the SCHe bottles. This pressure differential allows helium to preferentially flow from the normal process supply as long as its pressure remains above the pressure setpoint for the SCHe bottles. When general-service helium is not available, the SCHe bottle supply would not be affected (safety-class separation criteria) and would be utilized upon an SCIC system purge initiation. When the SCHe system is actuated, the two paths connected to the VPS line connected to the MCO long process tube provide helium to the MCO. The two paths connected to the VPS line connected to the MCO filtered process exit port provide a vent path. This feature is achieved by the differential pressure balance between the two system configurations and the pressure control valves in the pathway to the process vent. The two SCHe supply lines to the long process tube...
are set at staggered pressures so that helium is drawn first from one bottle and then the other. The two vent lines are set at equal pressures so that both provide helium for long-term pressurization. The return gas flow then backfeeds into the SCHe system lower pressure supply headers, through pressure regulators, and into the process bay local exhaust HVAC and process vent system. A 125 lb/in$^2$ safety-class rupture disk is provided for each SCHe line. The rupture disk is located between the pressure control valve and the check valve.

Under faulted conditions the pressure control arrangement provides purge flow through any single-line break. Under the condition of a pipe break between a safety-class isolation valve and the MCO, the flow of helium is as follows:

- **Process line break on the MCO long process tube side with little or no bulk water in the MCO**: Purge flow is through the MCO from the MCO HEPA filter supply side (the 4 lb/in$^2$ gauge helium purge flow is sufficient to overcome the differential pressure of the HEPA filter). MCO pressure is maintained slightly positive with flow of helium through the line break and/or out the process vent, depending on the size of the leak.

- **Process line break on the MCO long process tube side with bulk water in the MCO**: The water is pushed through the broken line until the pressure of the purge gas is overcome by the static head of the water column in the long process tube. At this point, the MCO is maintained at a positive pressure to preclude oxygen ingress.

- **Process line break on the MCO HEPA filter side with little or no water in the MCO to preclude air ingress**: Purge is through the MCO from the MCO long process tube side out the line break.

- **Process line break on the MCO HEPA filter side with bulk water in the MCO**: The SCHe system pressure overcomes the backpressure of the long process tube filled with water or the MCO is vented to allow hydrogen (at low initial concentrations) to disperse into the process bay.

The SCHe system interfaces include the general-service helium system, electrical and instrument air supply process bay local exhaust HVAC and process vent system, process hood support stand, MCS, building structure and SCIC system (the MCS cannot operate any of the SCHe isolation valves because all control functions come from the SCIC only).

**Safety Considerations**

The SCHe system provides safety functions for several of the design basis accidents described in Chapter B3. Local safety-class SCHe system bottle pressure gauges are provided at each SCHe system bottle, to provide indication of SCHe operability. Other instruments in the SCIC system which form part of the pressure boundary, are safety class for confinement purposes only.
The SCHe system isolation valves are designed to fail open, therefore, a loss of electrical power or instrument air will actuate the SCHe system purge and vent processes.

Sufficient purge flow volume and time under pressure is provided to allow time to initiate and complete recovery operations (e.g., isolation of the MCO at the shield plug valves). The "SCIC ISO & PURGE" trip alarm in the control room signals that the SCHe system has actuated.

A redundant relief path for the SCHe system is made up of safety-class regulators and 125 lb/in² gauge safety-class rupture disks. This relief path is for a high pressure failure caused by non-safety pressure regulators.

The SCHe system's functional requirements, performance criteria, and capability to perform the required safety functions are evaluated in more detail in Chapter B4.

B2.5.5 Condensate Collection Systems

Summary Description

Condensate from the process bay recirculation HVAC systems and the process general supply/exhaust HVAC system (condensate from the HVAC cooling coils) is routed via copper piping to condensate collection tanks. A condensate collection tank is hung underneath the mezzanine of each of the process bays containing a process bay recirculation HVAC system, and the condensate collection tank for the process general supply/exhaust HVAC system is located in the transfer corridor. The condensate collection tanks have sight glasses for visual inspection of liquid levels in the tanks, tank vents, and capped tank drains. Water in the collection tanks may be sampled and verified to determine appropriate disposition. Water can be transferred to the PWC system or drained via a flexible hose to an appropriate disposal container.

Safety Considerations

The condensate collection system is classified as general service.

B2.5.6 Effluent Drains Systems

Summary Description

Each process bay is equipped with an effluent collection system, with the major components of the system consisting of a floor sump and a drain. The process bay sumps are serviced by a sump drain pipe that leads to a retention basin. The function of the effluent drains system is to collect primarily fire suppression water in sumps in each of the five process bays from which they can be drained to a 20,000-gal below-grade retention basin located west of the CVDF.
In the case of a spill in any process bay, the sumps will hold up to 200 gal of liquid each.

The sumps are located in the center of each bay's floor. The floor is sloped toward the sumps to facilitate drainage in the case the fire suppression system operates. The sump drain pipe has a drain valve at each sump, and there is an inlet valve at the interface between sump drain pipe and the retention basin. These valves (which are normally closed) are manually opened except for situations involving fire. In case of actuation of the fire suppression system in a process bay, the sump drain valve for that specific bay automatically opens to allow the water flow of the fire sprinklers to pass through the sump and into the retention basin.

In the case of fire sprinkler actuation, water from the retention basin is sampled to determine an appropriate method of disposal. Water may be discharged from the retention basin to the ground. This disposal to the soil column is regulated under State Waste Discharge Permit ST 4508, which the Washington State Department of Ecology issued to the DOE Richland Operations Office. Rules governing the quantity and quality of water that can be discharged are described in SNFP Process Standard 409, Discharge to Ground.

Safety Considerations

The effluent drain system is classified as general service.

B2.5.7 Process Water Systems

This section discusses the systems that will treat, sample, and temporarily store process water drained from the MCOs in the CVDF.

B2.5.7.1 Process Water Conditioning System

Summary Description

The PWC system contains two receiver tanks (a total capacity of approximately 300 gal), pumps, filters, and ion exchangers that remove particulate matter and dissolved ions from the process water. The PWC system is designed to service one MCO at a time. The PWC system vacuum source is a water jet ejector that requires a water circulation loop. The PWC system continuously circulates water through the ejector, thus having a continuous source of vacuum for draining MCOs, VPS condenser tanks, the tempered water (annulus) system, or the cask-MCO annulus (unless the system is shut down for maintenance). The MCS controls all non-safety automatic functions and alarming of the PWC system. The SCIC has interlocks and trip control of the PWC-to-MCO isolation valves, provides a control room alarm for prepurge and postpurge failures, and a seismic trip to the PWC pumps. See Section B2.5.9.2 for more details.
Normal operations of the PWC system include draining water, processing collected water through the ion exchange modules (IXMs), and transferring processed water to a 5,000 gal facility storage tank (tank PWC-TK-4001 in the process water tank room). System valving can be used to direct flow via the following pathways: PWC system water recirculation from the receiver tanks through the ejector back to the receiver tanks, IXM flow back to the receiver tanks, and IXM flow through the filter to the PWC system holding tank. See Figure B2-17 for the process systems flow diagram. A layout of the process water tank room (showing the PWC skid, receiver tanks, particulate filter, samplers, IXMs, and storage tank) is provided in Figure B2-22.

The PWC system is powered by two (one running and one standby) motor-driven pumps capable of pumping 45 gal/min at a head of 140 ft of water. The ejector is capable of pulling at least 5 gal/min of liquid at a head of 17 ft of water. The pump discharges into a primary receiver tank through a water jet ejector. Operation of the PWC system vacuum pumping and receiving system includes control and monitoring of the pump operation, flow loop performance, receiver tank conditions, and suction line performance. Three-way flow control valves are aligned to operate one pump at a time.

The PWC system impurity removal section consists of a flow branch for feeding process water through one of the two IXMs (one is available on standby to allow for rerouting of process water if the other IXM must be isolated from operation). After cleanup, flow is then directed to a separate particulate filter prior to water transfer to the PWC storage tank. Automatic sampling is done before and after the water passes through the IXMs and after it passes through the particulate filter.

The IXMs are self-shielding, single-use, disposable water treatment units used for the removal of radionuclides from the water at the CVDF. The IXMs are self-contained units that consist of six carbon steel tanks connected to a common inlet and outlet. During IXM construction, the tanks are encased within a block of concrete. While the IXMs are in service, the concrete serves as shielding for internal radiation that emanates from the radionuclides entrapped within.

Control of process water to the IXMs is through a flow valve controlled by a flow-indicating controller. Normally, flow is continuously fed through the IXMs. High-differential pressure alarms are triggered if the IXMs begin to plug. Operating experience will determine what differential pressure to set. IXM loading with transuranics is administratively controlled as determined by sample analysis and likely requires IXM changeout prior to reaching the differential pressure limit. After the process water has passed through the IXMs, the treated water can be returned to the receiver tanks for recirculation through the water conditioning process or directed for final transfer of processed water to the facility storage tank.

An IXM typically remains in service until the differential pressure across the IXM reaches a setpoint or until administrative transuranic loading limits are reached. At that time, the IXM is disconnected from the CVDF piping and transported to a disposal location. Roof hatches are
incorporated into the CVDF design and an IXM can be changed out using a crane. Changeout of IXMs through the roof hatches was identified in the hazard analysis (HNF-SD-SNF-HIE-004) but not analyzed because IXM replacement is not expected during the lifetime of the facility (SNF-5197). The IXM will be monitored during initial process operations in order to trend filter loading. These data are necessary to establish IXM changeout requirements. Any potential future IXM removal and replacement will be analyzed for safety consequences.

PWC system operating limits and normal operating conditions are shown in Table B2-4.

<table>
<thead>
<tr>
<th>Table B2-4 Process Water Conditioning System Operating Limits and Conditions</th>
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<td><strong>Flow</strong></td>
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*This is an administrative limit based upon economic and regulatory constraints for transuranic material.*

The use of stainless steel PWC system piping and valves ensures that there will be negligible corrosion over the design life of the CVDF. A corrosion allowance of 0.06 in is specified in the ASME code data book for the VPS equipment.

**Safety Considerations**

The PWC isolation valves and associated piping are safety class. The PWC system has a safety-significant line to the receiver tanks and the remaining components are general service. The tanks in the IXMs are critically favorable by design. See Section B2 7 3 for additional details on criticality protection.
The PWC system filter and IXMs have shielding incorporated in their design.

The PWC system provides confinement of contaminated effluents in the process water, radiation protection from the effluents, worker safety hazards protection, criticality control, and natural phenomena hazard mitigation. The PWC tanks are inerted with helium from the general-service helium system before the MCO is drained to purge any air from the tanks and after draining to purge hydrogen from the tanks.

The water storage tank (PWC-TK-4001) is equipped with level indication and has an overflow pipe to allow for excess water in the storage tank to collect on the floor of the process water tank room.

**B2 5 7 2 Conditioned Water Shipping System**

**Summary Description**

The function of the conditioned water shipping system is to provide transportation of purified process water from the CVDF back to the K Basins integrated water treatment system for final treatment. The conditioned water shipping system consists of a pump (located in the process water tank room, pipe (from the process water tank room to spare bay 1), and flex hose to connect to a tanker truck. After process water storage tank sample results are verified, a tanker truck is backed into process bay 1 (room 131) and a transfer line is connected to the tanker truck using a flex line and quick disconnect. The tanker truck has a HEPA-filtered vent. Water is pumped from the PWC storage tank to the tanker truck using the storage tank pump. The water level in the storage tank is monitored.

**Safety Considerations**

The conditioned water shipping system is classified as general service. Any conditioned process water released from the tanker truck while it is in spare bay 1 is contained by the CVDF process bay sump. Specialty equipment is used to remove spills in the process bay to the PWC system.

**B2 5 7 3 Contaminated Water Sampling and Analysis System**

**Summary Description**

The contaminated water sampling and analysis (SMP) system is a subsystem of the PWC system. The SMP system is used to take water samples before process water is fed into the PWC system IXMs, after water exits the IXMs, and after water exits the particulate filter.

The SMP system contains sampling devices that are directly controlled by the MCS shielded collection bottles and instrumentation and controls that sample water from the process stream of the PWC system. See Figure B2-17 for the process systems flow diagram.
The collection bottles are periodically removed and sent to a laboratory for analysis. The results of the analysis determine the quantity of radionuclides removed from the MCOs, the radionuclides contained in the IXMs, and the water quality being transferred to the process water storage tank. The PWC impurity removal operation consists of a flow branch for feeding process water through IXMs and a particulate filter. Three samplers are provided and operate whenever flow is registered through the IXMs. The MCS controls the sample rate by sending a momentary sample initiation signal to the samplers.

**Safety Considerations**

The SMP system is classified as general service. The SMP system provides confinement of contaminated effluents from the process water, radiation protection from the effluents, and worker safety hazard protection.

**B2 5 8 Cooling Water Systems**

This section discusses the MCO tempered water (annulus) system and other associated water heating and cooling systems.

**B2 5 8 1 Tempered Water (Annulus) System**

**Summary Description**

The function of the tempered water (annulus) system is to maintain the MCO at the proper operating temperatures during processing. This includes heating the MCO before draining and cooling the MCO after processing.

The tempered water (annulus) system is a closed-loop system that provides tempered water (10 °C to 50 °C [50 °F to 122 °F] and 15 to 20 gal/min) to the annulus between the MCO and the cask (see Figures B2-17 and B2-23) and provides a means for controlling the temperature of the MCO contents. Pressure in the tempered water (annulus) system is approximately 20 to 35 lb/in² gauge. The tempered water is heated or cooled depending on the step in the drying process and is recirculated through the annulus with a water pump. The water level in the annulus is monitored by the SCIC system and the MCS on both supply piping and return piping.

Two water lines are connected to the cask-MCO, one near the bottom of the cask, which serves as the tempered water inlet, and one in the cask-MCO seal ring, which serves as the annulus outlet. The inlet line consists of a rigid, supported, vertical double-walled pipe with a double flex line lower section. The double flex line has a cask connection fitting that is mated to the cask-MCO. The upper end of the line connects to the process flange hood. Double containment is provided to an elevation above the top of the SNF contained within the MCO.
An inflatable seal-ring assembly forms part of the hydraulic boundary between the cask and the MCO. The cask–MCO seal ring is considered a part of the tempered water (annulus) system but is attached to the process hood. The seal ring’s function is to form a water- and air-tight seal in two locations: (1) between the cask shell and the seal ring on a horizontal plane and (2) between the MCO and the seal ring on a vertical plane. The center of the seal ring is open to allow process system connection access to the ports on the top of the MCO. Rubber seal ring bladders are inflated by the instrument air system after the process hood-seal ring assembly is bolted into place.

The tempered water is heated by an electric heater and chilled by the tempered water cooler. Chilled water to the cooler is supplied from the tempered water cooling system described in Section B2.5.8.2. The tempered water (annulus) system supply pressure to the cask–MCO annulus is controlled using a pressure control valve, and the pump discharge pressure is monitored.

A pump receiver tank is provided (TW-TK-3*12). The water level in this tank is monitored and has a high-level alarm.

**Safety Considerations**

The tempered water (annulus) system has three safety functions: (1) maintain water in the cask–MCO annulus, (2) maintain safe cask–MCO inlet water temperature, and (3) provide cask–MCO water level low-level alarm and indication. All of these functions are required throughout the cold vacuum drying process until the final pressure rebound test is completed.

The tempered water provides redundant safety-class temperature instrumentation on the inlet of the tempered water (annulus) system. The instrumentation trips the SCIC on high temperature and removes power from the tempered water heater. The water level in the annulus is monitored on the supply and return piping. The equipment design for maintaining the water level is passive; however, level alarms are provided to the SCIC system for annunciation in the control room if levels become low. The SCIC system has two independent temperature switches that will trip the tempered water heater when its setpoint is exceeded. The level alarm and temperature trip, which provide process parameter indication in the control room, are provided to the MCS. The annulus level alarm is annunciated on the SCIC annunciator panel in the control room.

The tempered water (annulus) system is designed to isolate the effects of failures on the non-safety portion of the piping. This includes line breaks, seismic activity damage, or process deviations. Water must be maintained in the annulus to provide a means to dissipate the heat generated in the MCO until the MCO is dried and passes its final pressure rebound test.

Recovery refill of the cask–MCO annulus is first attempted by use of a special tool (jumper between the deionized water line and the cask refill port). If deionized water is unavailable (the deionized water system is inoperable) to the tempered water (annulus) system, operators can refill...
the annulus manually with water, using a funnel, through a fill port and valve in the outlet line attached to the cask–MCO seal ring. A refill vent port is also provided to facilitate this manual refill operation by providing a visual indication of water level near the top of the supply line. Two parallel antisiphon valves are connected between the safety-class portion of the inlet and outlet piping of the deionized water system to prevent water from being inadvertently siphoned from the annulus in the event there is a line break in the piping.

The tempered water (annulus) system line from the connection flange on the process hood and extending to the lower port connection on the SNF cask is qualified as safety class B2 5 8 2 Tempered Water Cooling System

Summary Description

The tempered water cooling system provides cooling water to the tempered water (annulus) system cooling coils (see Figure B2-23). The cooling water provides the tempered water (annulus) system with the ability to cool the cask–MCO assembly during the MCO cooldown. The key process requirement for the tempered water cooling system is that it supplies cooling water of a designated temperature at a designated flow rate to the tempered water cooler.

The tempered water cooling system is used to supply cooling water at 10 °C (50 °F) and 20 gal/min to a tempered water (annulus) system cooler (see Figure B2-23). The system cools the tempered water recirculation loop during the cask–MCO cooling operational phase. The cooling water source is a closed water loop supplied by a heat exchanger and two pumps. The chilled water system removes the heat from the closed loop. The chilled water for the tempered water cooler is supplied by an intermediate cooling loop interfaced with the chilled water system through a process heat exchanger. The intermediate loop consists of a circulation loop with spare pumps. The cooling circuit provides cooling to the MCOs in each of the four process bays at any given time. Up to four MCOs can be processed, and the system is designed to remove 73 kW (250,000 Btu/h) (SNF-2356). The temperature of the water supply to the MCO annulus is controlled by the MCS except for the SCIC high temperature trip.

Pressure in the tempered water cooling system at 50 lb/in² gauge is greater than pressure in the tempered water (annulus) system at 20 to 35 lb/in² gauge. Pressures are higher in the cooling coils such that contamination spread into the cooling water system would be minimized during a postulated coil leak.

Safety Considerations

The tempered water cooling system is classified as general service.
B2 5 8 3 Chilled Water System

Summary Description

The chilled water system is designed to serve the cooling loads of the HVAC systems and the tempered water cooling system. The chilled water system includes packaged chiller units, circulating pumps, and associated piping, instrumentation, and controls. See Figure B2-24 for the layout of the major components of the chilled water system in the mechanical room.

The chilled water system serves the cooling coils in the process bay recirculation HVAC supply systems, the process general supply/exhaust HVAC system, and the process heat exchanger for tempered water cooling. The chilled water system has two fully redundant pumps. Two-way control valves are used on each cooling coil distribution loop. The chilled water refrigeration subsystem is comprised of two partially redundant, split-system, air-cooled water chillers with remote condensing units. Each chiller has a dedicated primary pump, piped in parallel to provide standby protection. The refrigerant system incorporates a chiller sequencing controller, staged compressors, bypass flow line, and back pressure valves to accommodate varying chilled water demand. Two chillers operate via a common sequencing panel and each system circulates chilled water at a constant volumetric rate. Check valves restrict backflow of water and a safety relief valve directs excess water through a drain line to the floor for collection via the effluent drain system.

Safety Considerations

The chilled water system is classified as general service.

B2 5 8 4 Vacuum Purge System Chilled Water System

Summary Description

The VPS chilled water system is used to supply chilled water at design temperature and flow rate to the VPS condenser and to the VPS condenser rundown tank cooling jacket (see Figure B2-25). A flow control valve controls the flow of water without MCS intervention. The chilled source is a closed, water-propylene glycol loop. The chilled water supply for the VPS condensers is a dedicated, low-temperature chiller (one chiller package for the four process bays with no redundancy). This chiller is capable of supplying water to all condensers. The water is supplied to the process skids via flexible hoses.

Safety Considerations

The VPS chilled water system is classified as general service.
B2 5 9 Instrumentation and Control Systems

This section discusses the instrumentation and control systems for the CVDF. These systems consist of the following:

- MCS
- SCIC
- Automatic temperature control (ATC) (the ATC is the control system for the CVDF HVAC systems. It is described in Section B2 6 2 1)

B2 5 9 1 Monitoring and Control System

Summary Description

The MCS is the interface between the operators and the CVDF process systems. Its function is to provide active indication, alarm, and control of MCO processing and support functions throughout the facility. The MCS provides automatic operation of each phase of MCO processing and accepts operator commands and overrides. The system provides process values and status to the operators and shift manager in a graphical format. The MCS also monitors the utility systems, including radiation monitoring, stack monitoring, VPS chilled water, tempered water cooling, helium supply, and instrument air. The MCS interfaces with the following process systems in each of the process bays:

- VPS
- Tempered water (annulus) system
- PWC system
- General-service helium system
- Deionized water system

In addition, the MCS receives signals from the SCIC system and, under normal circumstances, is allowed to control MCO isolation valves used by the SCIC system. However, the MCS cannot override SCIC system commands. The MCS does not support, nor is it required to operate for, the SCIC system to perform its required functions.

The MCS consists of interactive computers, logic circuits, panels, cabinets, input/output modules, and the wiring up to but not including process sensors (e.g., pressure, temperature, flow). There are four networked computers, including two operator workstations and an engineering workstation in the control room (room 107 in the administrative building), and one workstation (with view access only) in the shift manager's office (room 103 in the administrative building). The main PLC control panel also is located in the control room and includes a local alarm with silence button. All central processing units are connected on a local area network, which allows simultaneous access to the control system from multiple displays.
controlled networked input/output modules are conveniently located in each of the active processing bays, mechanical room, and the process water tank room. The field-mounted remote input/output modules can be individually controlled from the central control room.

**Safety Considerations**

The MCS is classified as general service.

**B2 5 9 2 Safety-Class Instrumentation and Control System**

**Summary Description**

The SCIC system provides actuation logic, actuation signals, and control interfaces. The SCIC system includes all of the equipment, actuation logic, electrical cabinets, conduits and wiring, local indication of the system status, and remote alarms and controls for system operation and testing.

The SCIC system provides active detection and response to process anomalies. Specifically, actuation of the SCIC system performs two safety-class functions: (1) signal closure of isolation valves leading to and from the MCO to the process systems and signal opening of SChE system isolation valves to provide MCO purging and pressurization, and (2) removal of power from the tempered water heater. The SCIC system also provides safety-class alarms in the CVDF control room for low annulus water level and notification of an MCO isolation and SChE purge trip. An additional PWC prepurge and postpurge failure alarm is also provided by the SCIC system as a defense-in-depth feature.

To perform these functions, the SCIC system monitors process and seismic parameters, detects off-normal conditions, and removes power from relays to de-energize:

- MCO isolation and SChE valves
- Tempered water heater
- PWC circulating pumps

To perform the annunciator functions, the SCIC system provides the following alarms:

- MCO isolation and purge
- Annulus low level
- PWC purge low (not safety-class but defense in depth)

The SCIC system requires no operator action to detect and respond to process upsets within the SCIC system controls. However, operator action is necessary to set the SCIC mode switch to the appropriate mode to facilitate operations.
The SCIC system has three distinct functions under defined upset conditions, or a manual trip.

**Multi-Canister Overpack Isolation and Purge** An MCO isolation and purge trip places the effected bay into the following conditions: VPS, general-service helium, PWC isolation valves closed (MCO is isolated) and SCHe valves open (MCO is purged). The following parameters can cause an MCO isolation and purge trip:

- MCO pressure (both positive pressure and vacuum)
- Helium flow rate
- Process bay high temperature switches
- Seismic trips
- Timer and vacuum trips

The MCO pressure and helium purge rates are used to prevent explosive accumulations of hydrogen and oxygen. MCO high pressure specifically provides for a safety-class vent path from the MCO. MCO pressure is also used for the 8-4-4 vacuum limit timer. The process bay high temperature is used to protect against the effects of high bay temperature on the accuracies of the safety-class instrumentations. Separate temperature switches are provided that, if exceeded, will initiate the safety-class isolation and SCHe system purge.

A vacuum limit timer monitors the time under vacuum to limit heatup of the fuel. This vacuum condition reduces thermal conduction from the MCO. After the vacuum cycle, a minimum time above atmospheric pressure is required to cool down the fuel. The SCIC system ensures that exceeding the time under vacuum, or returning to vacuum without first meeting the minimum time above atmospheric pressure, will result in MCO isolation and SCHe purge.

**Removal of Excessive Heat Input to a Multi-Canister Overpack** The tempered water high temperature trips remove power from the water heater through the heater contactors.

**Safety-Class Annunciation to Control Room** The SCIC system provides safety-class alarms to the control room (room 107) for low annulus water level and MCO isolation and purge trip. In addition, alarms for PWC prepurge and postpurge failure are provided as a defense-in-depth feature.

**System Configuration**

The SCIC system consists of two logic trains and redundant instruments and control functions. There is a set of SCIC panels located in each process bay. One of the two panels contains a PLC that provides one of the two logic trains for that process bay. The other PLC for that process bay is located in the adjacent process bay. Redundant instruments and control functions are provided to the two SCIC logic trains A and B. The PLC in process bay 2 serves as train A for process bays 2 and 3. The process bay 3 controller serves process bays 2 and 3 as
A similar arrangement is used for process bays 4 and 5. The PLC performs the various setpoint comparisons and provides safety-related timing functions.

Separation is achieved per Institute of Electrical and Electronics Engineering requirements identified in Table B4-3. Each process bay has instrument racks and SCIC instrument panels for monitoring and control of train A and train B for that process bay. The instrumentation consists of the following capabilities in each process bay:

- A local manual ISO & PURGE button that isolates the MCO and activates the SCHe system (activation of either train A or B ISO & PURGE pushbutton will initiate a trip that will close one set of safety-class MCO isolation valves to sufficiently isolate the MCO and will activate the SCHe system, both ISO & PURGE buttons must be activated to provide MCO isolation by both safety-class valves in each process system line)
- ISO & PURGE trip reset button
- Trip status and indicator lights for BAY ISO & PURGE trip, tempered water high temperature trip, seismic trip, process bay high temperature trip, and cask annulus low level alarm status
- Tempered water high temperature trip reset button
- Logic test switches

In a similar manner, the CVDF control room has SCIC system train A and B control room panels for each bay with the following capabilities:

- SCIC mode selection panel with duplicate key switches for the following operation sequence selection: (1) BYPASS, (2) HEATUP, (3) DRAIN, (4) PURGE/FLUSH, (5) DRYING, (6) PROOF, and (7) PRESSURE TEST; the two panels in the control room have key switches for all operations in process bays 2 through 5
- An annunciator and control panel with large lighted alarms for BAY ISO & PURGE, ANNULUS LOW LEVEL, and PWC LOW FLOW, alarm test capability, alarm acknowledgment, and annulus low level alarm bypass switches, and ISO & PURGE buttons for MCO isolation and activation of the SCHe system for each of the process bays.

During cold vacuum drying processing of an MCO, the SCIC system is placed into the mode appropriate for each processing stage as shown in Figures B2-10 and B2-11. When the MCO is first received, the SCIC system is positioned in the (1) BYPASS mode while the cask headspace is being vented, the cask lid is being removed, the process hood seal ring assembly is being bolted on, and the two sealing bladders are being inflated. The tempered water (annulus)
The SCIC system is connected. The SCIC system is switched to the (2) HEATUP mode when the port connectors are attached and the MCO headspace is being vented. When water is to be removed from the MCO, the SCIC system switch is placed on the (3) DRAIN mode. The SCIC system switch is then positioned in the (4) PURGE/FLUSH mode to flush particulate from the drain line. When the MCO is to be evacuated for vacuum drying, the SCIC system is placed in (5) DRYING mode, and when the drying is completed and proof testing is to be performed, the mode switch is placed in the (6) PROOF position. After proof checking, the SCIC system is placed in the (7) PRESSURE TEST mode for the one-hour pressure rebound test and preparing the MCO for shipment. As a final step, the MCO port valves are closed and the MCO is prepared for shipment and then the SCIC system is placed in the (1) BYPASS mode. Each of these modes enables the SCIC system to monitor different sensors with setpoints and logic appropriate for that processing step. Redundant sensors located in the process systems provide inputs to the SCIC system as indicated below:

- MCO pressure, from the VPS
- Helium purge flow rates, from the general-service helium system
- Tempered water high temperature switches, from the tempered water (annulus) system
- Cask-MCO annulus water low level switches, from the tempered water (annulus) system
- Process bay temperature switches
- Seismic recorder and trip status integral to the SCIC

The seismic monitors will activate the SCIC system trip independent of the logic. The seismic monitors directly trip the SCIC system output relay, which actuates the SCHe system and MCO isolation valves. A seismic event will also trip the tempered water high temperature protection circuit that shuts down the tempered water heater. Loss of electric power will de-energize the SCIC system output relay, which will isolate the MCO, actuate the SCHe system, and cut power to the tempered water (annulus) system heater. Once a seismic trip is activated all other logic is circumvented and no longer has a safety-class function.

SCIC PLC functional tests or calibrations require the use of an SCIC calibration and test computer unit that will test logic functions and allow for detailed calibration functions. The calibration and test computer is part of the SCIC system but is not normally connected to the SCIC system PLC. The safety-class cabinets are bolted to the process bay floor to meet performance category 3 seismic requirements.

To ensure sufficient reliability of the seismic system and to reduce spurious trips, three independent sensors are located in separate process bays. In addition, two-out-of-three trip logic
Annex B — Cold Vacuum Drying Facility

The SCIC system also provides safety-class alarms in the control room. The SCIC system monitors the tempered water (annulus) system low level switch and activates an alarm that notifies the operators when the water level in the cask-MCO annulus is low. In addition, the SCIC system also monitors the tempered water high temperature switch and initiates a tempered water (annulus) system heater shutdown if the temperature exceeds the setpoint.

A diagram of the SCIC system boundary (the various inputs and their associated responses) is provided in Figure B2-26. The general layout for the SCIC and seismic shutdown systems is provided in Figure B2-27.

Safety Considerations

The SCIC system is a safety-class system.

B2 6 CONFINEMENT SYSTEMS

This section identifies and describes the functional and physical and operational features of the confinement systems that provide protection against release of liquid and airborne radioactive or hazardous material. The primary confinement feature consists of the stainless steel MCOs. The MCOs are transported to the CVDF inside a transportation cask that protects the MCO. The mechanically closed cask-MCO combination forms a confinement barrier during transportation and receipt. The primary confinement of airborne and liquid effluents from the spent fuel during processing in the CVDF is provided by the MCO and isolation piping.

The process bays, the process water tank room, and surrounding rooms use HEPA-filtered exhaust systems to maintain appropriate building ventilation zone pressures such that air-flow is from less to more potentially contaminated areas. Secondary confinement is provided by the process bay local exhaust HVAC and process vent system and the building structures, in conjunction with the process general supply/exhaust HVAC system. The process general supply/exhaust HVAC system along with the process bay local exhaust HVAC and process vent system provide differential pressure for the CVDF process bay area, process support area, and process water tank room.

The administrative building has no confinement function and is ventilated for personnel comfort and air quality. The building is maintained at a positive pressure with respect to
atmosphere to preclude inadvertent air flow from process areas that are maintained at negative pressure

B2 6.1 Confinement Approach

The MCO and isolation piping of the cold vacuum drying processing systems form the primary confinement boundary during processing. The MCO is connected through three process lines to the VPS, general-service helium system, deionized water system, and PWC system process equipment. The pipes and valves that make up the isolation piping of the cold vacuum drying process systems form the physical boundary that isolates the MCO contents from the secondary confinement. All equipment exhausts and vents are directed through the process bay local exhaust HVAC and process vent system. Any leaks from the MCO or from the process equipment will be into the process bay or process water tank room.

The process bay local exhaust HVAC and process vent system provides secondary confinement at the top of the MCO with an open face process hood and filters all contaminated air flows from the process hoods, process vents from the VPS and tempered water (annulus) system, cask venting, and the SCHe system prior to discharge from the CVDF. During normal operation, secondary confinement is provided with a differential pressure established by the process general supply/exhaust HVAC system. Each process bay recirculating supply system is automatically controlled individually to adjust temperature and differential pressure of each bay by the ATC system.

The cask-MCO annulus is physically isolated from the MCO interior. The tempered water circulating in the cask-MCO annulus could be contaminated from the K Basins water. Therefore, the tempered water (annulus) system could also be contaminated. At the top of the MCO, a pressurized isolation barrier (MCO-cask seal ring assembly) between the cask and the MCO isolates the water in the cask annulus from the atmosphere in the process bay. The inflatable seals (barrier) are pressurized with a connection to the instrument air supply. There are two connections from the cask annulus to the tempered water (annulus) system. The cask-MCO forms a physical boundary for the cask annulus water. The tempered water (annulus) system pipes, valves, pumps, and heat exchangers form the physical boundary isolating the cask annulus water from the process bay. All equipment exhausts and vents are directed through the process bay local exhaust HVAC and process vent system. The system design isolates and controls releases of water from the cask annulus. Any releases from the cask annulus, or from the tempered water (annulus) system will be into the process bay confinement area (process bay local exhaust HVAC and process vent system, process bay recirculation HVAC system, process general supply/exhaust HVAC system, and/or floor sump and drain).

Contaminated process water removed from the MCO during processing is pumped to the process water receiver tank located in the process water tank room (room 132). Access to the room is via a change room (room 130). The process water tank room is maintained at a negative pressure via the process general supply/exhaust HVAC system. The piping from each process bay
to the receiver tank and the tank itself form a confinement boundary for the contaminated liquid. The piping is routed through each process bay to where it enters the process water tank room. All piping runs are within the confinement of the process bays or the process water tank room. The process water storage tank vent is HEPA filtered through a breather filter back to the process water tank room. The PWC system receiver tanks are vented to the process bay local exhaust HVAC and process vent system.

Liquid spills within each process bay will be confined in the sump in each process bay.

The process water tank room is designed as a liquid retention basin to contain all the water from the PWC system (e.g., tanks, piping, valves) in case of a leak or spill.

Liquid process confinement systems and the VPS were discussed in Section B2.5 HVAC confinement systems are discussed in the following subsections.

**B2.6.2 Heating, Ventilation, and Air Conditioning Systems**

Two HVAC systems (i.e., process bay local exhaust HVAC and process vent system and process general supply/exhaust HVAC system) provide airborne radioactive material confinement within the radiologically controlled areas of the CVDF and provide HEPA-filtered discharge via the CVDF stack. Each process bay also has an independent process bay recirculation HVAC system that provides outside air supply and HEPA-filtered recirculation for heating and air-conditioning. The humidity is monitored in all HEPA-filtered ventilation systems, and a humidity sensor alarms in the control room on high humidity. The CVDF administrative building has an independent supply and exhaust HVAC system that operates at a positive pressure to preclude inleakage from outside or adjacent areas. Figure B2-28 shows a composite flow diagram of the HVAC systems. A layout of the major components of the process bay recirculation HVAC system and process general supply/exhaust HVAC system is provided in Figure B2-29.

All exhaust systems are functioning during normal operation. Monitoring of the differential pressure facilitates maintaining confinement in the facility except when the telescoping door to a process bay is opened. If a telescoping door is open (e.g., when an MCO is being received or shipped out), the process bay recirculation HVAC system inlet damper is closed to increase flow through the doorway.

The HVAC systems utilize isolation dampers to stop the flow of air to maintain confinement within the facility and preclude cross-contamination between areas during upset conditions.
Process General Supply/Exhaust Heating, Ventilation, and Air Conditioning System

Isolation  The process general supply/exhaust HVAC system is isolated during any of the following events:

- Upon failure of the HVAC general supply fan, the general supply air isolation damper is hard-wired to close. The general exhaust fan continues to operate.

- If the HVAC general supply air temperature sensor detects a temperature of 4 °C (40 °F), then the general supply fan stops and the general supply air isolation damper closes. The general exhaust fan continues to operate.

- Upon detection of smoke at the supply air duct smoke detector, the general supply fan will turn off and the supply air isolation damper will close. The general exhaust fan continues to operate.

- The process general supply/exhaust HVAC system inlet isolation dampers will close on loss of power or instrument air. This is to preclude backflow out the inlet ducts which are not HEPA-filtered.

- The process general exhaust isolation dampers in each bay will close on a loss of power or instrument air. This is to maintain confinement in the process bays. These dampers also close upon a fire alarm in each bay.

Process Bay Recirculation Heating, Ventilation, and Air Conditioning System

Isolation  The process bay recirculation HVAC system is isolated during any of the following events:

- Upon sensing the personnel door or roll-up door position is open, the ATC closes the supply outside inlet air isolation damper for the affected process bay. Upon closure of either door, the outside inlet air isolation damper opens. The unaffected process bays continue to operate with no intervention.

- Upon detection of smoke at the supply duct smoke detector, the supply fan will turn off and the outside inlet air damper will close. The unaffected bays continue to operate. The general and local exhaust fans continue to operate with no intervention.

Process Bay Local Exhaust and Process Vent System Isolation  The process hood isolation dampers fail closed. The hood isolation dampers are provided with an instrument air reservoir for operation on standby power. Fire alarms in each bay will close the process hood isolation damper.
B2 6 2 1 Automatic Temperature Control System  The ATC system is capable of integrating multiple building functions including equipment supervision and control, alarm management, energy management, and historical data collection and archiving.

The ATC system is a control network composed of two levels of control (1) operator workstations and (2) application-specific PLCs. A local network utilizes a dual independent media network for critical data transfer and control between the redundant operator workstations and PLCs.

Two operator workstations are provided. One is in the control room (room 107) and one is in the mechanical room (room 207). The workstation in the mechanical room is the ATC system main control panel. The ATC system main control panel contains two PLCs (one primary and one secondary) that operate as redundant controllers. Only the primary PLC controls the I/O; however, if the primary PLC fails or is placed offline, the secondary PLC asserts control of the system without operator action to ensure continued system operation. The PLCs contain the system control program logic for all software interlocks, relationships, and actions. Control parameter manipulation for system operation is accessed via the operator workstations at either the control room (room 107) or the ATC system main control panel (room 207), or portable PC connections. Because all control is maintained at the PLC level, control is not dependent on either operator workstation for operation. The operator workstations enable control parameter manipulation, system monitoring, and data archiving.

A differential pressure alarm panel is located in the control room (room 107) where it provides continuous, independent monitoring of the differential pressure in each process bay (including process bay 1) and the process water tank room. A dedicated circuit from the facility uninterruptible power supply (UPS) panel (see Section B2 6 2 4 for discussion of the UPS system) supplies 120 VAC to the differential pressure alarm panel.

In the event of loss of normal power, the PLCs are powered by the UPS and undergo an orderly shutdown to prevent the loss of database or operating system software. Upon restoration of normal power, the PLCs automatically resume full operation without manual intervention.

Safety Considerations

The ATC system has been classified general service. The ATC system is relied upon for normal operation of the ventilation confinement systems. It is not required during operation of the process bay local exhaust HVAC and process vent system on standby power.

B2 6 2 2 Administrative Building Heating, Ventilation, and Air Conditioning System

Summary Description

The administrative building HVAC system maintains indoor space temperatures. The temperature is actively controlled by the administrative building ATC system. The system
maintains the space pressure of the administrative building positive with respect to change rooms, toilet rooms, the adjacent process areas of the facility, and the outside. The administrative building HVAC system uses two units: (1) a packaged heat pump unit located outdoors on a concrete mounting pad to the west of the building, and (2) an air conditioning and evaporation unit that supplies conditioned air to the electrical/communications room (room 108).

The heat pump air handling unit services all rooms within the administrative building excluding the electrical/communications room. Air is recirculated in this system with a minimum of makeup air to maintain air quality and provide air supply for the change room and shower area exhaust system (per ASHRAE-62). Air from the change room and shower areas is exhausted directly to the outside and is not recirculated. To prevent possible contamination, air in the corridor connecting the administrative building (corridor 115) to the transfer corridor of the process building (corridor 116) is not recirculated but is directed toward the transfer corridor (which is exhausted by the HEPA-filtered general exhaust system).

**Safety Considerations**

The administrative building HVAC system does not serve any safety functions and is classified as general service. Differential pressure monitoring and alarms for upset conditions are provided between the administrative building and the process area to ensure the correct direction of air flow. The control room is not required to be manned after DBAs, therefore, the supply ventilation is not provided with HEPA filtration.

**B2 6 2 3 Process Bay Recirculation Heating, Ventilation, and Air Conditioning System**

**Summary Description**

A majority of the process bay air is recirculated to conserve energy. The supply and return air passes through a prefilter and two HEPA filter stages to control airborne contamination. Makeup air is supplied, as required, to maintain air quality in the process bay and modulated to maintain differential pressure in the process bay. Each process bay air intake duct is provided with a pneumatic isolation damper, volume damper, and backdraft damper to prevent backflow of air to the outside. The outside air louver is provided with a mesh screen. The mix of outside air and interior air flow rates is controlled by dampers and can be adjusted to meet airflow and differential pressure requirements. Alarm and monitoring of all HVAC functions is provided by the ATC system in the control room and in the mechanical room.

The process bay recirculation HVAC system is sized to supply air to the process bays at a minimum rate of six air changes per hour. Outside air supply to maintain air quality is provided to each process bay. Normal process bay temperatures are maintained at a maximum of 25.5 °C (78 °F) in cooling mode and a minimum of 22.2 °C (72 °F) in heating mode. The temperature is actively controlled by the ATC system. Instrumentation is provided for detection of smoke in accordance with NFPA 90A, *Standard for the Installation of Air Conditioning and Ventilating Systems*, and for controlling space temperature. Equipment and ductwork that make up the...
process bay recirculation HVAC system is supported and anchored in accordance with the requirements necessary to meet the criteria of performance category 2 construction. All ductwork for the process bay recirculation HVAC system is fabricated of galvanized steel.

**Safety Considerations**

The process bay recirculation HVAC system is classified as general service. The pneumatic isolation damper at the air inlet is safety significant. The system continuously removes dust and airborne radioactive particulate, if present, from the air within the process bays, thereby limiting levels of airborne radioactive contamination within the process bays to acceptable levels during normal operations. The process bay recirculation HVAC system is designed and qualified for performance category 2, as defined in DOE-STD-1020-94. The process bay recirculation HVAC system does not discharge exhaust air to the environment. A backdraft damper is provided on the outside air inlet to prevent airflow reversal to the outside. The volume damper is modulated by the ATC system to maintain differential pressure in the process bays. The pneumatic isolation damper at the air inlet is safety significant and fails closed on loss of normal power. On standby power the process bay local exhaust maintains the differential pressure in the process bays.

**B2 6 2 4 Process Bay Local Exhaust Heating, Ventilation, and Air Conditioning and Process Vent System**

**Summary Description**

Each process bay contains a branch of the process bay local exhaust HVAC and process vent system. This system serves the MCO process hood and the process vent streams that may normally be contaminated (e.g., vacuum pump exhaust, VPS condenser tank vents, tempered water [annulus] system tank vents, SCHe discharges and the 30 lb/in² rupture disk).

The process bay local exhaust HVAC system is designed for the following functions:

- Maintain sufficient differential pressure in the process bays during normal operations to confine any airborne radionuclides within the CVDF ventilation systems and associated confinement structures. A differential pressure shall also be maintained following a loss of normal electrical power via support of the standby power system (see Section B2 8 5) when other ventilation systems are not operable. The system shall also prevent overpressurization of the exhaust ductwork following a loss of power, instrument air, or other offnormal conditions.

- Maintain a minimum dilution airflow rate to preclude potentially flammable concentrations of hydrogen within the process bay local exhaust HVAC system. A minimum flow rate of 1,000 ft³/min maintains a nonflammable concentration in the local exhaust duct.
Prevent accumulation and transportation of water that could migrate into other vent lines or damage equipment or components including HEPA filters.

Isolate the vent path from the cask vent or MCO vent at loss of electrical power. During cask or MCO venting, dilution of flammable mixtures transported to the exhaust duct is controlled by a flow switch located in the local exhaust duct. When the flow drops below the setpoint, the vent path is isolated via a solenoid valve from the duct. After reestablishing the flow rate, the vent path to the local exhaust can be opened.

Pneumatic isolation dampers are provided on the duct connecting to the process bay process hood. The process vents are open to the HEPA filters during operation and in the event the system shuts down. Two 100% capacity exhaust fans (located in the mechanical room) are provided for the process bay local exhaust HVAC and process vent system. The air in this system passes through an air handling unit (located in the mechanical equipment room) containing a prefilter and two HEPA filter stages before exhausting through the CVDF stack. Each stage of filtration is actively monitored for pressure drop through the use of pressure differential switches. The system has low flow switches on each process bay duct branch that provide alarm of system upset. Lead and standby local exhaust fans are monitored through individual pressure differential indicating switches. The exhaust fans' variable frequency motor drives provide constant volume with HEPA filter differential pressure increase and speed indication and general alarm inputs to the ATC system. If the lead local exhaust fan fails to start or fails during operation, an alarm is given at the control room and the stand-by local exhaust fan automatically starts. The ATC system is located in the CVDF control room and the mechanical room, where alarm and monitoring of all HVAC functions is provided.

A standby power system, conforming to the applicable general electric supply requirements of DOE Order 6430 1A and the NFPA 70, National Electrical Code, is available to support the CVDF hydrogen mitigation strategy. The standby power system supports the local exhaust system functions of maintaining adequate differential pressure for confinement in the process bays and reestablishing the minimum airflow rate as required for hydrogen dilution in the local exhaust system ductwork during a loss of normal electric power. The isolation dampers fail closed upon loss of electrical power, and each damper is provided an 8 3-gal compressed-air reservoir to operate the damper during standby power operation. Additional details on the standby power system is provided in Section B2 8.5.

The process bay local exhaust HVAC and process vent system process hoods effectively capture airborne contamination at its source by achieving a capture velocity of 125 ft/min at the top of the MCO in accordance with DOE Order 6430 1A and ACGIH Industrial Ventilation Manual hood design (ACGIH 1998).

Equipment and ductwork that make up the process bay local exhaust HVAC and process vent system are supported and anchored in accordance with the requirements necessary to meet...
the criteria of performance category 2 construction. All ductwork for this exhaust system is fabricated of stainless steel and is of welded flange construction.

Manual isolation dampers allow for isolation of individual exhaust fans. Exhaust fan discharge backdraft dampers are provided after the exhaust fans to prevent airflow reversal. Process hood isolation dampers fail closed.

The following process vent sources are routed as separate lines to the process bay local exhaust HVAC system ductwork because they are expected to be contaminated or contain potentially flammable concentrations of hydrogen, oxygen (air), or hydrogen and oxygen at various times:

- SCHe vent lines (expected to contain hydrogen)
- Tempered water tank vent line (expected to contain oxygen)
- Cask lid vent line (expected to contain both hydrogen and oxygen at different times)
- Vacuum pump discharge line (expected to contain both hydrogen and oxygen at different times)
- PWC receiver tank vent line (expected to contain both hydrogen and oxygen at different times)

**Safety Considerations**

The process bay local exhaust HVAC and process vent system process vent is classified as safety significant and is designed to provide sufficient airflow to dilute the potential hydrogen releases to nonflammable concentrations. This exhaust system provides this function for cold vacuum drying normal operation and designated accident conditions.

The process bay local exhaust HVAC and process vent system maintains a negative differential pressure in the process bays and captures and removes airborne contamination during normal operation and during certain mitigated accident scenarios described in Chapter B3. The system uses HEPA filters to remove the particulate contaminants from the exhaust stream prior to discharging the exhaust air to the environment via an exhaust stack.

**B2 6 2 5 Process General Supply/Exhaust Heating, Ventilation, and Air Conditioning System**

**Summary Description**

The process general supply/exhaust HVAC system is sized to supply air to the support confinement areas at a minimum rate of six air changes per hour. System capacity can maintain...
spaces at a maximum of 25.5 °C (78 °F) in cooling mode and a minimum of 22.2 °C (72 °F) in heating mode. Instrumentation is provided for detection of smoke in accordance with NFPA 90A and for controlling room temperatures. Room differential pressures are to be maintained at design values that will promote air flow within the CVDF from areas of lower potential contamination to areas of higher potential contamination. Pneumatic isolation dampers are provided on all duct branches connecting to the general process supply/exhaust HVAC system to prevent backflow in the event of exhaust system shutdown. Equipment and ductwork that make up the process general supply/exhaust HVAC system are supported and anchored in accordance with the requirements necessary to meet the criteria of performance category 2 construction. All ductwork for the process general supply/exhaust HVAC system is fabricated of galvanized steel.

The transfer corridor, associated support rooms, the process water tank room, and the mechanical room are supplied with conditioned air from the process general supply HVAC system. The air in this system is not recirculated. Pneumatic isolation dampers are provided on the outside air intake and tank room supply duct to prevent backflow of air in the event of system shutdown.

Air is also supplied to the process bays from the transfer corridor. Air enters into the process bays through transfer grills equipped with backdraft dampers from the process bay change rooms (rooms 118, 121, 124, 127, and 130). The backdraft dampers close on loss of air flow. Process general exhaust isolation dampers are provided at each process bay and fail closed upon facility loss of electrical power or loss of instrument air. Closure of the dampers enables the process bay local exhaust HVAC and process vent system operating on standby power to maintain differential pressure in the process bays and preclude backflow and communication among the process bays and the transfer corridor.

The process general supply/exhaust HVAC system provides exhaust air flow to maintain a negative pressure differential in the process bays and process water tank room with respect to all other areas external to the bays (the process water tank room has a lower differential pressure than the process bays). Air from the transfer corridor, associated support rooms, the process water tank room, the mechanical room, and the four process bays with process equipment skids is exhausted into the process general exhaust HVAC system. Fail closed pneumatic isolation dampers are provided on all duct branches connecting to the process general exhaust HVAC system to prevent backflow in the event of exhaust system shutdown. Two 100% capacity exhaust fans are provided for the process general exhaust HVAC system. The air in the system passes through an air handling unit in the mechanical room. The air handling unit contains a prefiltter and two HEPA filter stages that are used to filter the exhaust air before exhausting through the stack. Each stage of filtration is actively monitored for pressure drop through the use of differential pressure instruments. Lead and standby general exhaust fans are monitored through individual differential pressure indicating transmitters. The exhaust fans have variable frequency motor drives to provide constant air flow volume with HEPA filtration differential pressure increase speed indication and general alarm inputs to the control station. If the lead general exhaust fan fails to start or fails during operation, an alarm is given at the control station and the stand-by general exhaust fan automatically starts. The ATC system is located in the control room.
and the mechanical room where alarm and monitoring of all HVAC functions is provided. The airflow through the process general supply/exhaust HVAC system is normally uncontaminated. Only during upset conditions (i.e., accidents) would radioactive contamination be vented by the general exhaust system. Such conditions include releases into a process bay or the process water tank room. For these conditions, the process general supply/exhaust HVAC system mitigates releases from the CVDF.

Manual isolation dampers allow for isolation of individual exhaust fans. Exhaust fan discharge backdraft dampers are provided after the exhaust fans to prevent airflow reversal.

**Safety Considerations**

Portions of the process general supply/exhaust HVAC system are classified as safety significant for accident mitigation, as discussed in Chapter B3.0. Exhaust air is HEPA filtered to remove radioactive particulate contamination prior to being discharged to the environment via the exhaust stack.

**B2.6.2.6 Reference Air System**

**Summary Description**

The reference air system provides capability for continuous monitoring of the differential pressure, relative to atmospheric pressure, of various spaces throughout the CVDF. The reference air pipe header is fabricated of copper tubing. The ATC provides general service visual indication and alarms in the control room for out-of-tolerance differential pressure conditions throughout the facility. A differential pressure alarm in the control room provides safety-significant differential pressure indication and alarms for the process bays and process water tank room. Piping and instruments that make up the reference air system are supported in accordance with the requirements necessary to meet the criteria of performance category 2 construction.

The process bays and the PWC system tank room are maintained at a negative pressure with respect to atmosphere and to the remainder of the CVDF by the process bay local exhaust HVAC system and the process general supply/exhaust HVAC system with the differential pressure controlled by the ATC system. The reference air system consists of a series of static differential pressure indicating transmitters, located throughout the CVDF, that are interconnected by copper tubing. The reference pressure is measured by a static pressure sensor located above the administrative building roof. Pressure differential is indicated locally with magnehelic gauges in the process water tank room (room 132), in the transfer corridor and access rooms (corridor 116 and rooms 117, 118, 121, 123, 124, 126, 127, 129, and 130), in the process bays in the mechanical equipment room (room 207), and the administrative building access corridor (corridor 115). The ATC system monitors all differential air pressures.
Safety Considerations

The reference air system is classified as safety significant and provides differential pressure instrumentation relative to the atmosphere for various spaces throughout the facility. The differential pressure indicates whether the HVAC confinement systems are functioning properly.

B2.7 SAFETY SUPPORT SYSTEMS

This section identifies and describes the principal systems that perform safety support functions. The descriptions include the purpose, a summary description, principal components operations, control functions, and safety considerations.

B2.7.1 Monitoring Systems

There are two personnel monitoring systems: the radiation monitoring system, which consists of area radiation monitors (ARMs) and continuous air monitors (CAMs), and the room air quality monitoring system, which provides oxygen monitors. The stack monitoring system is also discussed below.

B2.7.1.1 Radiation Monitoring System

Summary Description

The radiation monitoring system provides radiological monitoring for radiation exposure levels and airborne radioactivity levels in order to identify personnel hazards, enhance worker safety, characterize workplace conditions, and verify effectiveness of physical design features and administrative controls. The radiation monitoring system conforms to the requirements of 10 CFR 835 and HSRCM-1, Hanford Site Radiological Control Manual. Additional CVDF radiological design requirements are provided by the following:

- The optimization principles as discussed in ICRP Publication 37, Cost-Benefit Analysis in the Optimization of Radiation Protection
- External exposure limitations identified by HSRCM-1
- ALARA principles (including the ALARA recommendations presented in NRC Regulatory Guide 8.8)

ARMs are provided to monitor medium- to high-energy gamma field levels within and outside the process bays and within the routinely occupied areas and restricted access areas in the CVDF. CAMs are provided to monitor alpha and beta-gamma radiation levels within and outside of the process bays and within routinely occupied or restricted access areas. Personnel
Contamination monitors are provided at strategic points to/from the process area for personnel monitoring and contamination control. ARMs (gamma) are located in each process bay (two per bay in bays 2 through 5, one in bay 1) and the process water tank room (one). The gamma monitor in the process water tank room is installed on or adjacent to the PWC receiver tanks to detect fissionable material buildup from the MCO.

Air sampling equipment is located in the vicinity of workers to provide an indication of the airborne radioactivity levels to which the workers are exposed. In addition to record samplers located near work stations, CAMs are provided to monitor airborne contamination levels within the CVDF to detect alpha, beta, and gamma radioactivity levels within routinely occupied and restricted access areas. Alpha CAMs are separate units from beta-gamma CAMs but are provided in pairs. Two pairs of CAMs (alpha and beta-gamma) are located in each process bay (one pair on the floor and one pair on the mezzanine), one pair in bay 1, one pair in the process water tank room, one pair in the transfer corridor, and one pair in the mechanical room (room 207).

Automated personnel contamination monitors are provided for personnel monitoring and contamination control in addition to portable instruments at commonly used egress points from the process areas. These devices include beta and gamma detection (with some capability for alpha). Six portal monitors are provided for personnel contamination monitoring. One walk-in monitor located at each process bay access vestibule including bay 1 (access to the process water tank room is controlled through the bay 1 process bay access vestibule), and one walk-through monitor located at the exit from the transfer corridor to the main change rooms (room 115).

Additional air monitoring for radioactivity is provided by fixed-head record samplers that are cart mounted and typically positioned adjacent to the CAMs. This equipment collects airborne particulate on a filter that is periodically removed for later analysis and replaced in accordance with radiological procedures.

All ARMs, CAMs, and personnel contamination monitors provide local indication of contamination levels. The ARMs and CAMs are also interfaced with the CVDF MCS. Remote indications and alarms will be displayed at the operations control room and a remote alarm in the radiological control technician room.

_Safety Considerations_

The radiation monitoring system is classified as a general-service system that is provided for protection of facility workers from exposure to radiation and radioactive contamination associated with operation of cold vacuum drying processes.
B2 7 1 2 Room Air Quality Monitoring System

Summary Description

The CVDF process bays and the PWC room (room 132) contain processing equipment. Because there is the possibility of introducing, through accident or equipment failure, inert gas (helium) into these rooms, it is possible that an oxygen deficient atmosphere might occur within these locations. Worker safety can be compromised if the oxygen content of the atmosphere within an enclosed room is deficient. The air quality monitoring system monitors the processing areas (bays 2 through 5) and the process water tank room for unsafe low-oxygen atmosphere content. Eight sensors are provided in each bay and four are located in the PWC room. There is one controller for each group of four sensors. Alarms are located in the transfer corridor adjacent to each process bay. Process water tank room alarms are located within the room and in the transfer corridor adjacent to room 130.

Safety Considerations

The room air quality monitoring system is classified as general service and provides for personnel protection against low oxygen atmospheres during cold vacuum drying processing.

B2 7 1 3 Stack Monitoring System

Summary Description

The stack monitoring system is a microprocessor-based particulate and iodine monitor and particulate and iodine collector mounted in a two-component, open-frame skid and cabinet assembly with a sample probe subassembly. The system consists of an active monitoring train and a record sampling train. The primary purpose of the record sampling train is to collect particulate and iodine effluent on fixed media for later laboratory analysis and documentation of emissions. The active monitoring train is designed to monitor real-time alpha and beta particulate and ¹²⁹I gamma radiation effluent in the CVDF HVAC ventilation exhaust stack. The iodine monitor is not required for safety purposes at the CVDF. It is part of a standard monitoring system purchased for use at the Hanford Site. The system provides warning of offnormal emissions. Visual digital and local alarm indications and contact output for remote alarms are provided for high radiation and common failure conditions. Analog and digital outputs are provided to the MCS. The following is provided by the stack monitoring system to the MCS:

- Stack flow rate
- Stack gas temperature
- Individual sampler and monitor flow rate and temperature
- Radiation indication for alpha, beta, and iodine
- High-radiation alarm (alpha, beta, or iodine)
- Common failure alarm
- Other monitor function indications and readouts as appropriate
A computer data logging system is provided to record monitored functions.

The stack monitor skid is located in the CVDF mechanical equipment room near the exhaust stack. Piping to and from the stack to the skid is routed in a structural steel support assembly. The system receives power from the facility UPS following loss of normal power (see Section B2.7.4 for a description of the UPS system).


A shrouded probe is used for stack sampling in accordance with the requirements of ANSI N13.1, Sampling and Monitoring Releases of Airborne Radioactive Substances from the Stacks and Ducts in Nuclear Facilities.

The stack monitoring system provides an online indication of alpha- and beta-emitting particles and gamma-emitting $^{129}$I, as well as continuous record sampling for radioactive particles. Standard, approved, record sampling techniques are provided to identify and determine the concentration of various isotopes (e.g., $^{238}$Pu, $^{239/240}$Pu, $^{241}$Pu, $^{241}$Am, $^{137}$Cs and $^{90}$Sr). The stack monitoring system meets the requirements of WAC 246-247, “Radiation Protection—Air Emissions,” DOE Order 6430 1A (Section 1589-99 01), and ANSI N13.1.


Safety Considerations

The stack monitoring system is considered general service and continuously monitors effluent releases to the environment through the CVDF stack. Alarms will initiate response actions, but these actions are not credited in Chapter B3 0 accident analyses.
Summary Description

The fire protection system provides fire detection, suppression, and loss limitation for the CVDF structure, personnel, and processing equipment.

The fire alarm system was designed and installed in accordance with NFPA 72, National Fire Alarm Code. Fire alarm signals are transmitted to the Hanford Fire Department via a radio fire alarm reporter. Annunciation of local building fire alarms and shutdown of appropriate HVAC supply fans are initiated upon receipt of a fire alarm. Manual pull stations are provided at emergency exits. Fire alarm annunciating devices (audible and visible) are provided for occupant notification.

The facility is automatically monitored for fire and smoke by ionization-type smoke detectors. Automatic sprinkler protection has been provided throughout the facility in accordance with NFPA 13, Installation of Sprinkler Systems. The operations support area is designed for ordinary hazard group 1 occupancy while the process bays and process bay support area are designed for ordinary hazard group 2 occupancy. Water is provided to the CVDF sprinkler system via a new 6-in line tapped off a new 8-in line looped around the facility. The loop is connected to the existing service water system at Building 165KW via an 8-in line. The water supply, as demonstrated by water flow testing, is adequate to meet the sprinkler system design flow and pressure demand plus a margin in excess of 10%. Fire hydrants are located just outside of the CVDF, one at the northwest corner and the other at the southeast corner. The facility design is such that no hose lay is greater than 275 ft.

All fire protection system valves, components, instrumentation, and controls required to perform fire detection and suppression functions are designated general service and are designed and qualified for seismic performance category 1 as defined in DOE-STD-1020-94. The pipe supports for those portions of the system that, upon failure, could impact the performance and function of safety-significant or safety-class equipment are designed and qualified for performance category 2, as defined in DOE-STD-1020-94. Portable fire extinguishers of the appropriate size and class are located in the CVDF in accordance with the criteria of NFPA 10, Standard for Portable Fire Extinguishers.

Safety Considerations

The CVDF is of noncombustible construction except for the built-up roof over the process bay area. The installed roof assembly is a membrane roof with 4 in. of rigid insulation that is adhered to the welded steel plate deck. The roof is considered by the fire hazard analysis, SNF-4268, Fire Hazard Analysis for the Cold Vacuum Drying Facility, to be a combustible roof that will support combustion. The wall between the administrative building and the process support area is a 2-hour fire-rated wall. The wall between the administrative building and process bay has fire resistance equivalent to a 2-hour fire-rated wall. The construction specification for
the interior finish meets the flame spread and smoke development criteria as required in NFPA 101 and the floors meet the critical radiation flux criterion required by NFPA 101.

The HVAC system is provided with fire screens and rate-compensated thermal detectors ahead of the filter housings. The HVAC system is designed to shut down upon detection of a fire. Upon receipt of an alarm from the detectors, the system closes the appropriate isolation dampers to that process bay and shuts down the respective process bay recirculation HVAC system fan.

Deviations from fire protection criteria are identified in Chapter B110 and in the fire hazard analysis (SNF-4268). Suggested actions for the deviations are documented in SNF-4942, *Spent Nuclear Fuel Cold Vacuum Drying Facility Implementation Plan for Fire Hazard Analysis Suggested Actions*.

### B2 7.3 Criticality Protection

The cask-MCO physical configuration has been analyzed to assure criticality control is accomplished by passive geometry control. Refer to Chapter B60 for additional discussion.

The process water removed from the MCO may contain fissionable materials from metallic uranium corrosion. The analysis in Chapter B60 shows that for all normal operations and for accident conditions that satisfy the double contingency criterion, the equipment in the CVDF handling the process water from the MCO meets the acceptance criteria when the limits and guidelines are met. The limits, controls, and engineering features are further described in Chapter B60.

The designed criticality prevention features in the CVDF are the receiver tanks and the IXMs in the process water tank room. This is the only equipment where fissionable materials could accumulate in the facility in significant quantities. The tanks and the IXMs are critically favorable by design, in that their physical geometries preclude a criticality event even though the amount of fissionable material that could accumulate in the equipment is too small to present a criticality hazard.

### B2 8 UTILITY DISTRIBUTION SYSTEMS

#### B2 8.1 Electrical Power Distribution System

*Summary Description*

The electrical power distribution system provides electrical power distribution for the CVDF, equipment, and instruments as required. Normal power is provided with a grounding system to ensure safety to personnel and equipment, to provide a connection to earth for
transformer neutrals, to provide a discharge path to ground through lightning and surge arresters, and to provide a reference point for electronic systems.

The electrical power distribution system design includes the extension of an overhead 13.8-kV primary circuit from existing poles at a point adjacent to the CVDF site. The existing circuit is extended and routed overhead to a point near the south edge of the construction site. Lightning arresters and fused cutouts are provided on the last pole to provide power to a 1,500 kVA, 13.9 kV/480Y-277 volt pad-mounted transformer via an underground feeder installed in the concrete encased duct-bank. The primary underground feeder cable is rated at 15 kV. Also, outdoor type potential transformers, current transformers, and the metering cabinet are provided at the last pole for the energy monitoring and providing input to the electrical utilities field data acquisition system. A simplified one-line diagram of the electrical power distribution system is provided in Figure B2-30.

All underground primary conduit is concrete encased and buried a minimum of 2 ft below finished grade. The backfill is done in 6-in layers with each layer compacted to 95% of maximum density. A conduit marking tape is provided above the underground conduit route at a depth of 1 ft below finished grade.

A less-flammable oil filled, 13,800-480Y/277 V, 1,500 kVA, pad-mounted, self-cooled, transformer is located on the southwest side of the CVDF. The transformer is in accordance with requirements from IEEE C57, IEEE Standards Collection Distribution Power and Regulating Transformers, and is provided with backup current limiting fuses, no-load tap changer, pressure control valve, and devices to indicate oil temperature, pressure, and liquid level. Secondary power of the transformer is routed to a 480-V switchboard located inside the CVDF electrical room (part of office area) through underground feeders installed in a duct-bank.

Power is distributed by a free-standing, metal-enclosed switchboard. Phase, neutral, and ground bussing are provided. Switchboard main and feeder circuit breakers have adjustable solid-state trip units responsive to long-time, short-time, instantaneous, and ground-fault current characteristics. Power from the switchboard is distributed to 480 V distribution panels located in each bay, a motor control center in the mechanical room, 75 kVA distribution transformer in the electrical room, 30 kVA UPS transformer, and selected HVAC loads.

Motor loads that are not part of prepackaged equipment are powered from a 480-V, 3-phase motor control center located in the mechanical room. The motor control center is freestanding and metal enclosed with modular plug-in combination motor controllers. Assembly and wiring complies with the National Electrical Manufacturer's Association ICS Standards Type II-C and Underwriters Laboratories standards. All wiring is extended to terminal compartments for ready access. Controllers are National Electrical Manufacturer's Association-rated.

Panel boards for power distribution, lighting, receptacles, and small loads are constructed in accordance with National Electrical Manufacturers Association and Underwriters Laboratories.
Main circuit breakers are provided in panel boards. Panel board circuit breakers are the molded case bolt-on type.

120-V duplex receptacles are provided around the inside perimeter of the process bays. Ground-fault circuit interrupting type receptacles are provided for the receptacles located within 5 ft, 0 in. of wet surfaces and sinks, as required by NFPA 70. Ground-fault circuit interrupting circuit breakers are provided for receptacles located in bays 1 through 5 and the tank room because of the decontamination activities in those areas. The administrative building is provided with 120-V duplex receptacles. Appropriate receptacles are provided for cord-connected equipment such as lunchroom equipment, computer equipment, and radiation monitoring equipment.

The design provides for installation of feeders (conduit and wire) adequately sized for equipment to be installed. Generally, all power and control conduits used within the facility are rigid galvanized steel. Where equipment packages (e.g., the process equipment skid) have several motors or electrical loads, not all of which will operate simultaneously, the feed has been sized for the loads expected to operate simultaneously. The electrical system components are designated performance category 2.

Safety Considerations

The CVDF electrical power distribution system is designated as general service. The electrical power distribution system interfaces with safety systems to provide normal power. Safety SSCs fail safe upon loss of normal power. The safety-significant standby power system is provided for the operation of essential loads in accordance with Section B2 8.5.

B2 8.2 Lighting (External/Internal/Exit/Emergency)

Summary Description

Exterior building lighting is provided. Metal-halide, wall-mounted fixtures with photocells are installed at all exterior doors.

Interior facility lighting consists of energy-efficient fixtures typically operated at 120 V. High-bay area lighting consists of metal-halide fixtures when ceiling height permits good lighting design. Lighting for storage areas and mechanical areas are high-output industrial fluorescent fixtures. Other interior lighting consists of surface- or flush-mounted commercial fluorescent fixtures.

Exit signs are provided at the interior of each building exit. Exit signs include integral chargers, batteries, and relays to provide illumination automatically upon failure of the normal power source.
System emergency lighting is provided inside the CVDF in accordance with NFPA 101. Emergency lighting in areas with fluorescent lighting is provided by use of fluorescent fixtures with integrally mounted backup battery packs with a 90-minute capacity.

**Safety Considerations**

The lighting is classified as general service.

**B2 8 3 Ground System (Power/Instrumentation and Control)**

**Summary Description**

Grounding characteristics at the Hanford Site are such that supplemental cable for ground grid or loops and attachment to well casings is required to satisfactorily reduce ground resistance. All underground metallic piping, all building structural steel and rebar, and all underground metallic conduit and ground cables at the building are tied into the building ground grid.

Equipment grounding conductors are provided in accordance with Article 250-57(b) of NFPA 70 (Table 250-95). A minimum of a No 12 American Wire Gauge copper ground conductor is used with low-voltage power circuits. Grounding conductors are sized as a minimum in accordance with NFPA 70 and IEEE-142, *Practice for Grounding of Industrial and Commercial Power Systems*.

All structural steel and mechanical equipment with integral electric motors or equipment is grounded. A grounding electrode system is constructed in accordance with NFPA 70, Article 250. Interconnected elements include building reinforcing steel, water piping, and a concrete-encased grounding electrode. A separate equipment ground conductor is routed with all power conductors and lighting circuits in accordance with NFPA 70, Article 250. A separate instrumentation and control ground system is provided in accordance with NFPA 70, Article 250.

**Safety Considerations**

The ground system is classified as general service.

**B2 8 4 Uninterruptible Power Supply**

**Summary Description**

A central facility UPS system is provided for the radiation monitoring instrumentation, process bay differential pressure instrumentation, oxygen monitoring system, special purpose receptacles, stack sampling, and for the MCS. This UPS system is supplied from a central
isolation transformer for clean power to these systems. A second UPS system is provided for the security system to provide continuous power during but not after a design basis accident.

The facility UPS system is a 30-kVA, 480 V to 120/208V UPS system. The facility UPS system is complete with transformer, battery charger, battery packs, battery disconnect switch and circuit breaker, manual bypass switch, and a maintenance bypass circuit. The facility UPS system UPS-1 (see Figure B2-30, Panel/UPS-1), located in room 129, mechanical equipment area, is provided to all MCS PLCs, process bay remote input/output panels, operator control stations A and B, Hanford Local Area Network receptacles, CAMs, ARMs, and oxygen monitors. The UPS system provides a minimum of one-hour run time at full load.

One 3.1 kVA, 208/120 V to 208/120 V, UPS system complete with battery packs, battery disconnect switch and circuit breaker, and maintenance bypass circuit, located in the process support area (room 126), is provided to the security system for continuous power supplied during but not after a design basis accident. All security sensors, cameras, and sensor and video transmission equipment is provided with a UPS system with a minimum of eight hours run time at full load.

There are no safety function requirements to provide UPS power to any of the SCIC panels or equipment (including the annunciator panel in the control room). To aid in the control room response to a loss of site power, the two SCIC annunciators do have UPSs to provide 30 minutes of power to aid in the transition from the normal state to a loss of power state. The seismic monitors have battery power which is normally being charged with normal site power. On loss of power, the seismic equipment will continue to operate to record valuable data for restart. This feature is not required for safety.

Other battery backup power is provided for specific systems only by use of direct-current battery systems and battery chargers. Battery backup systems are provided for the following:

- Fire alarm control panel (mounted adjacent to the panel)
- Emergency egress and exit illumination (integrally mounted)
- Communications system (integrally mounted)

Safety Considerations

The facility UPS is classified as general service.
B2 8.5 Standby Power System

Summary Description

The CVDF utilizes a standby power system that conforms to the applicable general electric supply requirements of DOE Order 6430 1A and NFPA 70. The standby power system supports the following process bay local exhaust HVAC system functions:

- Reestablishing the minimum airflow rate as required for hydrogen dilution in the local exhaust system ductwork during a loss of normal electric power.
- Maintaining adequate differential pressure in the process bays to confine airborne radionuclides within the CVDF ventilation systems and associated confinement structures following loss of normal electrical power.

The CVDF standby power system consists of the following major components:

- A 100-kW diesel generator with an air-start motor (with integral air-operated relay valve and electrically operated pilot valve) and battery to operate engine controls, a second 100-kW diesel engine is in the diesel generator building but is not connected to the facility.
- A generator including an automatic voltage regulator, main generator circuit breaker, and current transformers.
- A automatic transfer switch that performs the following functions:
  - Selects either normal power or standby power for supplied loads.
  - Provides an engine start signal if the normal source voltage drops below a set level for a duration in excess of a selected delay.
  - Disconnects the load bank from the generator upon transfer to the standby power source.
- A load bank to test the diesel generator.
- A 50-gal capacity day tank providing a local source of fuel for the diesel engine.
- A 500-gal diesel storage tank.
- An air receiver that provides a source of stored energy to start the diesel engine.
- An air compressor to maintain pressure in the air receiver.
A load bank circuit breaker that provides overcurrent protection between the generator output circuit breaker and the load bank

A battery charger to maintain the diesel engine battery

The standby power system components (e.g., diesel engine, generator) are located in a 36 ft by 18 ft by 13 ft high building. The building is located 37 ft north and 100 ft west of the CVDF.

The standby power system provides power to the following systems:

- Process bay local exhaust HVAC system and restart circuit (Section B2 6 2 4)
- Process bay heat trace (Section B2 5 3 1 2)
- Facility UPS system (Section B2 8 4)
- Instrument air compressor (Section B2 9 2)

A simplified electrical system one-line diagram of the standby power distribution is provided in Figure B2-31.

The diesel generator components of the standby power system are designed to performance category 2 criteria. No redundant generator or dual feed capabilities are required. The standby power system is designed to restart the process bay local exhaust HVAC system to reestablish the required minimum dilution airflow rate following loss of normal electrical power to the local exhaust fans. Startup of the standby power system occurs upon loss of normal power to the automatic transfer switch for a duration in excess of a selected time delay. The standby power system has sufficient fuel to provide power for 24 hours.

**Safety Considerations**

The standby power system is classified as safety significant. The standby power system supports the restart of the process bay local exhaust HVAC system following loss of normal electrical power. Restart of the process bay local exhaust HVAC system minimizes the accumulation of potential flammable hydrogen concentrations within the process bay local exhaust HVAC system ductwork and maintains a negative differential pressure in the process bays.

**B2 8 6 Lightning/Surge Protection System**

**Summary Description**

Lightning protection is provided for the CVDF using pole-line mounted lightning arresters. The lightning protection system complies with all requirements of NFPA 780 and NFPA 70. The lightning/surge protection system is passive. All system equipment installed on the upper 25 ft of the exhaust stack is lead-covered copper or stainless steel. Air terminals and down conductors.
are provided on opposite sides of the stack, and an encircling conductor provided at both the top and bottom of the stack

**Safety Considerations**

The lightning protection system is classified as general service.

The lightning/surge protection system includes surge arresters on the 138 kV power pole where the overhead is converted to underground cable to protect the service transformer from switching and lightning-induced surges.

The facility stack includes a lightning protection system consisting of stack-mounted air terminals (rods and copper downcomer wires) connected to the ground system.

### B2.9 AUXILIARY SYSTEMS AND SUPPORT FACILITIES

#### B2.9.1 Communications Systems

**Summary Description**

The communications system provides operations personnel the ability to communicate within the facility for the efficient operation of the facility systems. The communication system interfaces with the 100 K Area communications, the Hanford Local Area Network system, and the local telephone network. The CVDF communications system consists of a voice paging system, a standard telephone system, and a data communications system.

The communications system is designed to send and receive routine CVDF operations messages between the CVDF administrative locations and the respective process and utility areas of the CVDF. In addition, the system is capable of transmitting and receiving emergency warning signals and messages to and from the 100 K Area alarm station. The CVDF communication system is designed in accordance with HNF-S-0403 *Standard Specification for Hanford Site Telecommunications Systems and Facilities.*

The voice paging system is controlled from the operations control room and is capable of one-way voice paging to wall-mounted speakers in the process bays, process water tank room, transfer corridor, mechanical equipment room, administrative building, and external equipment pads. Wall-mounted speakers are located throughout the CVDF. The voice paging system is capable of sending public address messages and alarms to multiple remote locations. Codes for evacuation and take-cover signals are included in the system code list. The voice paging system also is capable of receiving and translating alarm initiation codes from the 100 K Area alarm station. The voice paging system is capable of recognizing the emergency signal codes and broadcasting the standard Hanford Site alarms over the system throughout the CVDF.
remote alarm station at the CVDF is located in the control room. The sound level produced by the speakers is at least 10 dB above the local background noise, and flashing lights are used in areas where needed.

The telephone system provides standard telephone service with outside lines to the CVDF administrative building, including the control room. A telefax machine is available. An intercom and voice paging capability is integrated with the standard telephone, and is controlled from the control room. Every station is capable of receiving voice paging. Each intercom station can be individually addressed from the control room. The intercom system consists of speaker phones located in each of the five process bays, the tank room, mechanical equipment room, transfer corridor, administrative building (e.g. men’s and women’s change rooms, radiological control technician and radiation monitor room, control room, and corridor), and various locations outside. In addition to the standard telephone service, a crash-alarm telephone is located in the control room to receive sitewide and other emergency messages. The process bay telephones can directly connect to the crash-alarm system for relaying emergency messages to operations personnel.

The data communications system consists of connections to the Hanford Local Area Network service in the administrative building (including the control room). The Hanford Local Area Network provides a variety of data services including e-mail, online controlled documents, reference materials, data processing applications, and Internet connections. This service is provided to standard computers with network cards and Hanford Local Area Network connections available in the standard telephone outlets that have outside line connections.

Safety Considerations

While the communications system provides information during emergency situations of significance to personnel safety, the system has no direct safety function at the CVDF. The communication system is classified as general service.

B2 9 2 Instrument Air Systems

Summary Description

The instrument air system is used to produce and distribute compressed instrument quality air throughout the CVDF. The instrument air system provides the instrument air used in the cold vacuum drying process, in the HVAC dampers, and for tools, and it provides the instrument air used to operate valves, process instruments and HVAC instruments.

The instrument air system provides dry, filtered, oil-free air at approximately 100 lb/in². The system has a specified rating of 251 ft³/min at 175 lb/in² gauge with a maximum dewpoint of -40 °C (-40 °F). The CVDF instrument air system runs continuously in an automatic mode and provides a 4-hour instrument air supply in the event of compressor downtime. The instrument air
system uses redundant duplex, two-stage, piston-type compressors capable of delivering a minimum of 25 standard ft³/min each at 175 lb/in² gauge. The system has a 250-gal receiver tank. Drying of the air is provided by an instrument air refrigerant dryer and a heatless air dryer with automatic regenerative desiccant beds capable of drying 35 standard ft³/min with housings capable of an operating pressure of 175 lb/in² gauge and 38 °C (100 °F) to -40 °C (-40 °F) pressure dew point. A coalescing filter and prefilters and after-filters (sized for a minimum of 35 standard ft³/min with housings capable of an operating pressure of 175 lb/in² gauge at 100 °F) remove particulate from the air and include differential pressure gages. The instrument air system also consists of a flex connector, pressure control valves, low-pressure alarm, check valves and associated piping. Condensate from the air compressor and refrigerant air compressor is directed to the floor drains. Figure B2-32 shows the layout of the instrument air system.

The instrument air system is used to operate the automatic spring return gas-operated valves in the process and safety systems, to inflate the MCO seal ring pneumatic seals, to purge the cask–MCO annulus following completion of the cold vacuum drying process, to operate pneumatic tools, and to operate the air ride suspension for cask trailer leveling. The instrument air system provides instrument air for the HVAC system solenoid-operated valves, damper position controls for isolation, volume, and backdraft. The system also provides approximately 25 lb/in² gauge air to the HVAC system for monitoring and control and to process instruments used for monitoring and control. The instrument air is routed to the work areas through a distribution system with valves and quick-connect fittings.

To ensure the process hood dampers on the process bay local exhaust HVAC and process vent system are reopened when standby power is activated and the local exhaust fan restarts, a safety-significant instrument air system is installed in each air line to each process bay hood isolation damper. The safety-significant portion of the instrument air system in each process bay consists of an air tank, double-check valves in the inlet air supply line to each instrument air tank, piping from the tank to the hood isolation damper actuator, and a pressure-indicating gauge on the tank. Each instrument air tank reservoir has a minimum capacity of 8.3 gal and is designed for a maximum pressure of 200 lb/in² gauge. The normal operating pressure of 100 lb/in² gauge provides sufficient air volume in the reservoir to complete four actuations of the isolation damper. Double-check valves prevent loss of air to possible line breaks in the general-service portions of the system and local air pressure indication is provided for the operator to monitor the tank pressure in each bay.

The instrument air system interfaces with several safety-class systems by providing pneumatic power to operate safety-class valves. These valves are designed to fail safe upon loss of air. The instrument air compressors are supplied with power from the standby diesel generator following loss of normal electrical power (see Section B2 8.5 for a description of the standby power system).
**Safety Considerations**

Even though several safety-class systems use instrument air to operate control or isolation valves, their fail-safe configuration upon loss of air eliminates any safety functions for the instrument air systems. The instrument air system is classified as general service.

Safety-class Instrument air filters are provided in conjunction with the safety-class gas operated valves.

The instrument air reservoirs in each bay are safety significant.

**B2 9 3 Cranes and Hoists**

**Summary Description**

Four overhead trolley cranes, one in each process bay, are used to remove and install the cask lid from the cask and to install and remove the process hood from the top of the cask-MCO. The cranes can be used to assist in maintenance activities within the process bay when required. The cranes are designed in accordance with natural phenomena and hazard performance category 1 criteria and are qualified for seismic 3-over-1 criteria. Seismic 3-over-1 criteria are described in Chapter B4.0. The CVDF cranes are designed to category 3 criteria to not fall into the process bays. All lifts involving cask-MCO equipment are classified as critical lifts.

The CVDF cranes have service ratings of 4,000 lb. The cranes are designed in accordance with the criteria included in CMAA 74, with class D heavy service rating. The cranes are top-running, single-girder with capping channel. The bridge, trolley, and hoist are driven by electric motors and are capable of operation from the mezzanine level by the use of a pushbutton station. The pushbutton station is a three-motion, eight-button unit with a National Electrical Manufacturers Association rating of 12. Controls allow for hoist up or down, carrier forward or reverse, trolley forward or reverse, and crane power on or off. The pushbutton unit is permanently mounted to the mezzanine handrail. The crane has brakes in accordance with CMAA 74 and limit switches to prevent overtravel of the hoist in the raise direction only. The crane has attached seismic stops that prevent the end trucks from lifting off the runway rail during seismic events.

**Safety Considerations**

The process bay cranes are considered general service, but are considered in the 3-over-1 assessment within the process bays. Safety considerations involve following approved procedures (e.g., required load testing) for performing critical lifts. Because the lifts performed by these cranes (i.e., removal of the cask lids) have the potential for damage to the MCO, the definition from 10 CFR 72.3, "Definitions," as implemented by HNF-SD-SNF-DB-003 classifies the cranes to be important to safety. However, because any damage resulting from a cask lid drop has been
analyzed to be very minor, the important-to-safety category C classification has been assigned. A technical safety requirement control is imposed to restrict crane movement during MCO processing except as part of an approved recovery procedure.

**B2 9.4 Service Water System**

*Summary Description*

Service water is supplied to the CVDF from the existing 100 K Area service water system on the west side of the 165-KW Building. Fire sprinkler service is supplied from the facility service water. Reduced pressure backflow preventers are installed at interface locations. All service water piping is designed and constructed in accordance with ASME B31.3 *Process Piping Code*, or ASME B31.9, *General Services Piping Code*.

*Safety Considerations*

The service water system is classified as general service.

**B2 9.5 Deionized Water System**

*Summary Description*

The deionized water system provides deionized water to the PWC system, to the MCO for flushing the connectors, and to the tempered water (annulus) system.

Deionized water is supplied to the process areas via a package deionized water unit. The deionization system is sized for a peak flow of 10 gal/min and an average load of 1,500 gal/day. The system consists of exchange units, one activated carbon filter to remove organics from the water feed, three mixed-bed deionizer tanks, piped in parallel with one online at any one time, a holding tank, a 20-gal pressure tank and a final 1-micron rated filter. The holding tank has a minimum capacity of 150 gal. Water received from deionized water tanks will be recirculated through the holding tank on a continuous basis.

*Safety Considerations*

The deionized water system is classified as general service.
B2 9.6 Potable Water System

Summary Description

Potable water is used to supply the sinks, rest rooms, custodial service sinks, drinking fountains, decontamination shower, the deionized water system, and makeup water for the chilled and cooling water systems. Hot water is supplied by tank-type electric water heaters. The safety eye washes are provided with portable water containers.

Potable water is supplied to the CVDF from the existing 100 K Area sanitary water loop on the west side of Building 165-KW. The potable water service main is a 2-in.-diameter copper pipe. Backflow preventers and pressure-reducing valves are inside the CVDF. Potable water feeds the administrative building plumbing fixtures in standard copper piping. The potable water also feeds two residential-type, 52-gal hot water heaters to provide adequate hot water for the four showers (hot potable water). The potable water supplies the process deionized water unit via a reduced pressure backflow preventer and the chilled water makeup connection. All potable water lines are insulated.

Safety Considerations

The potable water system is classified as general service. Backflow preventers required to prevent backflow of water from process areas are considered industrial safety and are general service.

B2 9.7 Sanitary Collection System

Summary Description

The CVDF is not served by a sitewide sanitary sewage system. Sanitary sewage generated in the CVDF is collected in a holding tank outside the CVDF. There are no connections to the sanitary sewage system in the process bays of the CVDF. The sanitary sewer plumbing is designed in accordance to the Uniform Plumbing Code (UPC 1988), and the tank is designed, installed, and will be operated in accordance with WAC 246-272, "On-site Sewage System."

Sanitary sewage from the CVDF is gravity drained to a 5,000-gal underground holding tank located west of the facility. The holding tank is equipped with a fluid level sensor. The pipe material is polyvinyl chloride or acrylonitrile-butadiene-styrene, except where exposed, where it is galvanized. The holding tank has the capacity to store three days of sanitary waste with maximum occupancy of the facility. The tank is expected to be pumped by site forces using a pumper truck designed for such purposes. A radiation monitor is provided as required by DOE Order 6430.1A
Safety Considerations

The sanitary sewer and the monitoring system are classified as general service.

B2 98 Security System

Summary Description

This system design description addresses the CVDF security system. The system’s primary purpose is to provide reasonable assurance that breaches of security boundaries are detected and that assessment information is provided to protective personnel. In addition, the system is utilized by Operations to support reduced personnel radiation goals and to provide reasonable assurance that only authorized personnel are allowed to enter and exit the designated security areas.

Radiation protection features have been incorporated into the design of the CVDF to reduce radiation exposures to security equipment and personnel consistent with ALARA objectives. The principal source of radiation hazards in the facility is from SNF when a loaded cask–MCO is present in any of the process bays. Security personnel do not enter the process bays as part of their normal job requirements. There are no specific hazards to security equipment located in the process bays from radiation exposure. No unique ALARA requirements or worker radiological exposure issues are identified for security personnel or the security system.

There are no unique or specific industrial safety hazards or criticality safety requirements associated with the operation of the security system.

Natural phenomena hazards include seismic events, high winds, tornadoes, flooding, lightning strikes, and loads from accumulation of snow or volcanic ash. Natural hazards were evaluated for the design of the CVDF. There are no design requirements for the CVDF security system traceable to the natural phenomena hazard evaluation.

There are no chemical or process design requirements for the CVDF security system.

There are no technical safety requirement required surveillance’s that apply to the CVDF security system. The security system fabrication quality assurance/control program is based on the application of a graded approach as described in HNF-MP-599, Project Hanford Quality Assurance Program Description.

Safety Considerations

All security system components are categorized as general service and are designed and qualified as performance category 1 per DOE Order 5480 28 and administered using the guidance of HNF-PRO-097.
The security system is a stand-alone dedicated system and does not interface with safety-related alarms (e.g., fire, radiation). Equipping all pedestrian doors with panic hardware for emergency egress accommodates personnel safety.

The security system components, installation, and testing meet the applicable requirements of NFPA 70.

**B2 9 9 Specialty Equipment**

**Summary Description**

Several items of specialty equipment are required to operate the CVDF. Summary descriptions are provided below:

- A truck and tanker will be used to transport conditioned process water from the CVDF to the K Basins integrated water treatment system.
- A shielded cask will be used to store and transport water samples from the PWC system for analysis.
- Swipe count pigs are supplied in each process bay for measuring contamination levels on the MCO and cask.
- A jumper in each process bay will be used to refill the MCO cask annulus with deionized water for emergency recovery purposes.
- Wheel chocks are supplied in each process bay for chocking the cask trailer wheels.
- A cask lid lift fixture is provided for each process bay to remove the cask lid before installation of the process hood and seal ring.
- A storage fixture is provided for each process bay to store the cask lid after removing it from the cask. Features for the fixture allow for the decontamination of the lid and for lid seal replacement.
- A special system (exhaust trunk) is installed in each process bay to collect the exhaust fumes from the tractor to prevent them from entering the process bay during delivery or removal of the cask trailer.
- A pneumatic hose with quick-connect on one end and glad hand and control valve compatible with the cask transport trailer is provided for each process bay. The hose connects to the facility's instrument air system.
A process hood lift fixture is supplied in each process bay to install and remove the process hood-seal ring to the cask.

A lifting fixture is supplied for the PWC tank room hatch.

A portable crane is supplied for the removal and installation of the IXMs.

A portable radiation monitor is supplied to monitor the process bay floor drains for radiation.

A portable calibration cart is supplied for the periodic calibration of the RGMs.

A storage locker is supplied to each process bay for the storage of various tools and equipment.

Various process flexible jumpers.

**Safety Considerations**

All of the specialty equipment has been classified as general service.

**B2 9 10 Special Tools**

**Summary Description**

Several special tools are required to operate the equipment in the CVDF. The following list is not intended to be all inclusive, nor does it identify all necessary components for maintenance of the process skids or facility. Summary descriptions of special tools required to operate the process are provided.

- A torque wrench with extension, torque multiplier, and sockets for the cask lid bolts is provided for each bay. Various socket sizes, wrenches, and pneumatic tools are available for use in each bay.

- Tools are provided for each bay to remove and install the cask drain port cover and to install the tempered water lower cask port connection.

- A tool is supplied for each bay to remove and replace the cask vent port plug.

- A tool is supplied for each bay to remove and install the MCO process port covers.

- A torque wrench and sockets are supplied for each bay to remove and install the MCO process port connector and to open and close the MCO port plugs.
Tools, other than standard, are provided for the maintenance of the PWC system, VPS, and tempered water (annulus) system

Tools, other than standard, are provided for the removal of the HEPA filters from the filter housings

Calibration tools are supplied to verify and recalibrate instrumentation

Leak check equipment for the MCO process port connectors and port covers

MCO vent port valve and pressure monitor

Cask annulus refill tools

Cask annulus refill water storage container, pump, and hand truck

Cask vent jumper tool

MCO vent jumper tool

Safety Considerations

Two items of specialty tools, the cask vent jumper tool and the MCO vent jumper tool, have been classified as safety significant and are discussed in Chapter B4.0. All other specialty equipment tools are classified as general service.

B2 10 REFERENCES


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Annex B — Cold Vacuum Drying Facility


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Figure B2-2 Side View of the Cold Vacuum Drying Facility
Figure B2-3  Typical Multi-Canister Overpack and Cask Package
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Annex B — Cold Vacuum Drying Facility

Figure B2-4  N Reactor Mark IV Fuel Assembly

Figure B2-4  N Reactor Mark IV Fuel Assembly
Figure B2-6  Cold Vacuum Drying Facility Architectural Second Floor Plan

NOTES
1. FOR NOTES AND LEGEND SEE B4-1-REV01
2. FOR WALL SPACES SEE B1-1-REV01
3. INDICATE "INDICATE" B5 IN 1/8" (20MM) HIGH
4. ROUTE 22 & 4 SIMILAR TO 22

FLAG NOTES
- HIGH VOLTAGE EXHAUST FANS & FILTERS
- OUTLET OF DRY FLOOR DOOR AND DOOR LOCK
- DRIER FILTER IN PLATFORM ACCESS
- EMERGENCY Dye High Station
- OUTLET OF CRANE HOIST TRAVEL LIMIT
- CHAIN LOCK
- EXHAUST ACCESS LADDER
- CONTINUOUS AIR MONITOR
- AREA INHABITATION MONITOR

ROOM DESCRIPTION

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Figure B2-9  Generalized Description of Water Removal from a Multi-Canister Overpack

Free Water
(Bulk liquid, liquid adhering to surfaces, liquid trapped in fuel crevices, liquid in particulate interstitial regions)

Hydration Water
(In oxides)

Chemisorbed Water
(Surface adsorbed)

Removed as bulk water by draining at the CVDF

Removed by the CVDF drying process

Residual water after cold vacuum drying < 200 g

CVDF = Cold Vacuum Drying Facility
Figure B2-10 Cold Vacuum Drying Process Block Diagram

1. MCO Transport Receipt
2. Cask Venting and Lid Removal
3. Process Hood/Seal Ring Installation and Tempered Water Connections
4. VPS Connection and Venting
5. MCO Heatup and Drain
6. Vacuum/Purge Drying Operation
7. No Purge Vacuum Check*
8. Pressure Rebound Pre-Test*
9. Pressure Rebound Test*
10. Proof-of-Dryness Mode
11. Second Pressure Rebound Test
12. MCO Cooldown
13. MCO Helium Leak Test and Prep for Cask Departure

*Failure to successfully complete these tests results in repeating the vacuum purge drying cycle of Step 6
Figure B2-11: Progression of Water during Removal from a Multi-Canister Overpack
Figure B2-13 Cold Vacuum Drying Facility System
Interfaces with a Multi-Canister Overpack

The bold lines identify the safety class portion of the MCO interface
Figure B2-14  Process Hood and Seal Ring Installation on the Multi-Canister Overpack
Figure B2-15  Typical Multi-Canister Overpack

SECTION A-A
- PROCESS VALVE
- COVER PLATE ASSEMBLY
- LOCKING AND LIFTING RING
- SET SCREW
- SEAL
- BASKET STABILIZER EXTENSION
- CANISTER COLLAR
- GUARD PLATE RING
- GUARD PLATE
- INTERNAL HEPA FILTER ASSEMBLY

SECTION B-B
- RUPTURE DISK BODY
- HOLE COVER PLATE
- SHIELD PLUG SUBASSEMBLY
- SEALS

SECTION C-C
- CANISTER COLLAR
- BASKET STABILIZER EXTENSION
- INTERNAL HEPA FILTER ASSEMBLY
- GUARD PLATE RING
- GUARD PLATE

SECTION D-D
- GUARD PLATE RING
- INTERNAL HEPA FILTER ASSEMBLY
- GUARD PLATE

BOTTOM PLATE SUBASSEMBLY
- PROCESS TUBE GUIDE CONE
- BASKET SUPPORT PLATE
- SHELL BOTTOM

MULTI-CANISTER OVERPACK ASSEMBLY
- FUEL/SCRAP BASKET
- CANISTER COLLAR RING
- GUARD PLATE RING
- SHELL
Figure B2-16  Shield Plug Subassembly
Figure B2-20  General-Service Helium Delivery to the Process Bays
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Figure B2-26  Safety-Class Instrumentation and Control System Boundary

Responses
- Isolate MCO
- Actuate SCHe System
- Cut TWS Heater and Circulation Pump
- Alarm for Annulus Low Water Level
- Alarm for Manual Corrective Action for PWC Receiving Tanks
- SCHe Blowdown After Drain
- PWC Circulating Pumps

Inputs
- Loss of Electricity
- Seismic
- Bay Temperature
- TWS Outlet Temperature
- Manual
- MCO Pressure (High and Low)
- MCO Helium Flow
- Annulus Water Level
- PWC Pre and Post Purge Failure
- Seismic Trip

Programmable Logic Controller

SCIC System Boundary

Legend:
- MCO = multi-container overpack
- PWC = process water conditioning
- SCHe = safety-class helium
- SCIC = safety-class instrumentation and control
- TWS = tempered water (annulus) system

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Figure B2-31 Electrical System One-Line Diagram of the Standby Power Distribution
Figure B2-32 Instrument Air System (sheet 1 of 2)
Figure B2-32  Instrument Air System (sheet 2 of 2)
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CHAPTER B3 0

HAZARD AND ACCIDENT ANALYSIS
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B3 4 ACCIDENT ANALYSIS
   B3 4 1 Methodology
   B3 4 2 Design Basis Accidents
      B3 4 2 1 Gaseous Release
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      B3 4 2 3 Multi-Canister Overpack External Hydrogen Explosion
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B3-19 Dose Consequence Summary for High-Pressure Overpressurization Release

B3-20 Source Terms for Releases from Mitigated Scenarios

B3-21 Total Dose Summary from Mitigated Overpressurization Scenarios at Several Receptor Locations

B3-22 Summary of Safety Features Required to Mitigate or Prevent an Overpressurization Reaction
LIST OF TERMS

ALARA as low as reasonably achievable
ARF airborne release fraction
ARR airborne release rate
BDBA beyond design basis accident
CSB Canister Storage Building
CVDF Cold Vacuum Drying Facility
DBA design basis accident
DOE U S Department of Energy
FSAR final safety analysis report
HEPA high-efficiency particulate air (filter)
HVAC heating, ventilation, and air conditioning
MAR material at risk
MCO multi-canister overpack
NRC U S Nuclear Regulatory Commission
PWC process water conditioning
RF respirable fraction
SCHe safety-class helium
SCIC safety-class instrumentation and control
SSC structure, system, and component
SNF spent nuclear fuel
TSR technical safety requirement
8-4-4 8-hour initial vacuum cycle, 4-hour subsequent vacuum cycles, 4-hour return to pressure between vacuum cycles
B3 0 HAZARD AND ACCIDENT ANALYSES

B3 1 INTRODUCTION

This chapter presents a summary of the methodology, assumptions, and results of the safety hazard analysis and design basis accident (DBA) analyses that have been performed for the Cold Vacuum Drying Facility (CVDF). These analyses form a Safety Basis for the final safety analysis report (FSAR) and present a comprehensive evaluation of the drying process and related activities at the CVDF, and the natural phenomena and external hazards that can affect the public workers, and environment. Single and multiple initiating events from equipment and human error failures in the facility have been considered, as well as human-generated and natural events (i.e., common mode failure) outside the facility. When the FSAR is approved by the U.S. Department of Energy (DOE), this Safety Basis will help to establish an Authorization Basis for the Spent Nuclear Fuel (SNF) Project CVDF. Changes to the facility during operations will be reviewed to ensure they do not affect the Authorization Basis. This review process is described in Chapter 17 of the SNF Project FSAR and is termed the unreviewed safety question process.

The contents of this chapter are as follows:

- The requirements for establishing the safety basis for the CVDF are listed in Section B3 2. The requirements listed consist of DOE orders and standards and applicable U.S. Nuclear Regulatory Commission (NRC) rules and guidance.

- The CVDF hazard analysis methodology and results are summarized in Section B3 3. The complete hazard analysis is contained in HNF-SD-SNF-HIE-004 Cold Vacuum Drying Facility Hazard Analysis Report. The hazard analysis identifies hazard sources, hazardous conditions, potential accident scenarios and their initiators, and preliminary assessments of event frequencies and consequences. Hazards are identified by form and location and represent a complete spectrum of events that could occur throughout the facility. An initial set of safety features that would serve to prevent or mitigate the postulated accident scenarios was identified during the hazard analysis. A final set of safety features is identified in the accident analyses in Section B3 4 2 and in Chapters B4 0 and B5 0. A safety feature that prevents an accident scenario is defined as one that reduces the expected annual frequency of occurrence for the accident beyond extremely unlikely (less than 10⁻⁶ per year). The hazard analysis has been revised to reflect the final set of safety features.

- The final facility hazard classification, determined in accordance with DOE-STD-1027-92, Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480 23 Nuclear Safety Analysis Reports is addressed in Section B3 3 2 2. The CVDF has been assigned a final designation of hazard category 2 nuclear facility.
Section B3.2.3 contains discussions of defense in depth, worker safety, and environmental protection, including a detailed tabulation of engineered and administrative features that have been identified as providing for worker safety.

The hazard evaluation ranking performed in the hazard analysis identifies hazards and associated events that pose a challenge to offsite and onsite radiological dose evaluation guidelines. This ranking is used to select unique and representative accidents, with sufficiently high risk estimates for further detailed quantitative evaluation as DBAs. SNF-2770, Cold Vacuum Drying Facility Design Basis Accident Analysis Documentation, documents the derivation of the DBA frequency range. The six major DBA categories are evaluated in the following sections:

- Section B3.4.2.1, “Gaseous Release”
- Section B3.4.2.2, “Liquid Release”
- Section B3.4.2.3, “Multi-Canister Overpack External Hydrogen Explosion”
- Section B3.4.2.4, “Multi-Canister Overpack Internal Hydrogen Explosion”
- Section B3.4.2.5, “Multi-Canister Overpack Thermal Runaway Reaction”
- Section B3.4.2.6, “Multi-Canister Overpack Overpressurization”

The analysis process resulted in identification of safety-class and safety-significant structures, systems, and components (SSCs), and the functional requirements needed to prevent or mitigate potential accident sequences and ensure that the facility can be operated in a safe, controlled manner. The analysis also identified a set of controls (i.e., technical safety requirements [TSRs]) that ensure all identified vulnerabilities associated with facility operation have been adequately addressed. When required by the SNF Project’s commitments to meet equivalent NRC requirements, candidate features have been identified as “important to safety” and designed, engineered, and procured consistent with safety-class or safety-significant classification requirements.

Each of the six analyzed DBAs represents a bounding case for a category of hazards and accidents. An in-depth review has been performed of all remaining accidents identified by the hazard analysis in each of the categories. The accidents reviewed typically represent the same accidents with different initiators, or different accidents bounded by the cases presented. This review resulted in the selection of additional safety-class and safety-significant SSCs and TSRS to ensure that all individual hazards and accidents within a category have been addressed. The table that accompanies each DBA in Section B3.4.2 includes the preventive and mitigative features and associated TSRS for the bounding case presented and for all other events within that accident category. Defense-in-depth features also are identified in these tables.
This chapter interfaces with several other SNF Project safety documents. This chapter identifies and evaluates all hazards associated with the CVDF and analyzes bounding accident scenarios associated with processes and activities that occur within the CVDF. The hazards associated with transport and the accident scenarios that are postulated to occur during shipment are analyzed in HNF-SD-TP-SARP-017, Safety Analysis Report for Packaging, Onsite Multi-Canister Overpack Cask. The planned scope and content of various other SNF Project safety reports, TSRs, and supporting safety documents are defined in HNF-SD-SNF-PLN-012, Spent Nuclear Fuel Project Integrated Safety Management Plan. In particular, MCO design basis information is provided in HNF-SD-SNF-SARR-005, Multi-Canister Overpack Topical Report.

The CVDF hazard and accident analysis relies on the SNF in a multi-canister overpack (MCO) to be clean (i.e., loose surface contaminants removed) and the fuel loading to be controlled (e.g., the mass of fuel in a scrap basket, the number of scrap baskets) at the K Basins. In addition, to protect the limiting values assumed in the hazard analysis and the DBA analyses, the interface between the CVDF and the Canister Storage Building (CSB) requires that an MCO delivered to the CSB contain less than 200 g of free water and provide confinement of the combustible hydrogen gases within the mechanically sealed MCO. Safety-related performance documentation from the K Basins and the CSB is relied on to ensure that the as-received and as-shipped content and condition of the MCO are “as required” for the safety basis properties of the MCO.

### B3 2 REQUIREMENTS

Chapter 3 of the SNF Project FSAR lists the design codes, standards, regulations, and DOE orders that contain requirements and guidance for establishing the safety basis for the SNF Project. Only the requirements that are specific for the CVDF and that pertain to the safety analysis are provided here:

- **DOE Order 5480 23, Nuclear Safety Analysis Reports**  
  DOE Order 5480 23, Section 8b (3)(e) and (k), in conjunction with its Attachment 1, "Interim Guidance for DOE Order 5480 23," sets the requirements for analysis. This chapter complies with these requirements by documenting performance of the hazard and accident analyses. The methodology and criteria used to identify facility hazards, hazard rankings, candidate accidents, DBAs, preventive and mitigative features and controls, and the classification of these features (along with the definition of safety functions, performance criteria, and applicability) are described in Chapter 3 of the SNF Project FSAR and are documented in this Chapter B3 0.

- **DOE Order 5480 22, Technical Safety Requirements**  
  This order sets the requirements for the development and preparation of a TSR document. This chapter complies with these requirements by documenting the performance of hazard and accident analyses using implementing guidance from HNF-PRO-704, Hazard and
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**Accident Analysis Process**, and DOE Order 5480 23  The results of the analyses have been used to identify specific safety functions of SSCs, performance requirements for the SSCs, and the times for application of the safety functions

- **DOE Order 6430 1A, General Design Criteria**  This order provides requirements for the identification of safety-class items  The analyses documented in this chapter use the SSC classification requirements of DOE Order 6430 1A to identify safety-class SSCs and their safety functions

**DOE-STD-1027-92 and DOE-STD-3009-94, Preparation Guide for US Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports**, are standards which provided guidance to satisfy DOE requirements of DOE Order 5480 23 in the CVDF hazard and accident analysis process and the preparation of this report

In Letter 95-SFD-167, Implementation of K Basins Spent Nuclear Fuel Project (SNFP) Regulatory Policy (Sellers 1995), DOE established the requirement for new SNF Project facilities to achieve "nuclear safety equivalency" to comparable NRC-licensed facilities  The SNF Project identified the NRC requirements that were to be met, in addition to existing and applicable DOE requirements, in order to establish nuclear safety equivalency  These NRC requirements and the process used to identify them are documented in HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward Additional NRC Requirements, and in WHC-SD-SNF-DB-010, Cold Vacuum Drying System Natural Phenomena Hazards  These documents establish the design requirements to be met for the CVDF to achieve NRC equivalency and considered the following

- **Title 10, Code of Federal Regulations, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste"** (10 CFR 72)  This rule is used for the licensing of independent spent fuel storage installations  Section 72 122, "Overall Requirements," requires that the design bases for SSCs important to safety reflect appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena  Section 72 24, "Content of Application Technical Information," provides requirements in Paragraph 72 24(m) for the analyses of accidents and natural phenomena events that could result in a dose at the controlled area boundary

- **NRC Regulatory Guide 3 26, Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants**  This guide establishes the format and content for license applications for fuel reprocessing facilities

While these documents have particular significance to this chapter, they do not, by themselves, establish requirements for the CVDF or the Chapter B3 0 accident analyses  Requirements identified in HNF-SD-SNF-DB-003 and WHC-SD-SNF-DB-010 address NRC equivalency  These requirements include safety documentation for (1) determining “important-to-safety” classifications for preventive and mitigative features, (2) identifying and resolving worker safety issues in DOE Order 6430 1A, DOE Order 5480 23, and

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DOE-STD-3009-94, and (3) evaluating nearby activities identified in Section B1 6, “External Human-Generated Threats,” that may represent a threat to the facility

Important-to-safety SSCs have been identified in accordance with 10 CFR 72 3 Once SSCs have been identified as having a function meeting the definition of important to safety, the requirements specified in 10 CFR 72 are imposed A graded approach is applied to an important-to-safety SSC by using the guidance provided in NUREG/CR-6407, Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety, as follows

- **Category A — Critical to Safe Operation**

  SSCs in Category A include those whose failure or malfunction could directly result in a condition adverse to public health and safety Important-to-safety SSCs in this category are classified as “safety class,” as defined in DOE Order 6430 1A, with the additional requirements therein

- **Category B — Major Impact on Safety**

  SSCs in Category B include those whose failure or malfunction could result in a condition adversely affecting collocated worker health and safety Note that from the definition of Category C, Category B is understood to include events that could significantly damage the MCO without severe impact to public health and safety SSCs in this category are classified as “safety significant,” as defined in DOE-STD-3009-94

- **Category C — Minor Impact on Safety**

  SSCs whose failure or malfunction would not significantly reduce containment and would not be likely to create a situation adversely affecting public or collocated workers' health and safety SSCs in Category C are classified as “safety class 3” (nonsafety), as defined in DOE Order 6430 1A

The determination of important to safety is directly incorporated into the tables of identified preventive or mitigative safety function features at the end of each DBA scenario in Section B3 4 2, and external human-generated threats are discussed in Section B3 3 2 3 The requirements of WHC-SD-SNF-DB-010 are incorporated into the natural phenomena hazard design criteria and are considered in the hazard evaluation

Letter 97-SFD-172, Risk Evaluation Guidelines (REGs) to Ensure Inherently Safer Designs (Sellers 1997) provides another SNF Project requirement that is significant in the development of this chapter This letter provides the onsite risk evaluation guidelines and offsite release limits for comparison to DBA consequences
B3 3 HAZARD ANALYSIS

The hazard identification and evaluation process provides a thorough, predominantly qualitative evaluation of the spectrum of risk to the public, workers, and the environment caused by accidents that involve the hazards identified in the hazard analysis. The hazard analysis process is fully described in Section 3.3 of the SNF Project FSAR. A final hazard analysis has been performed to support the accident analyses and is documented in HNF-SD-SNF-HIE-004. The final hazard analysis systematically reviewed the final CVDF design, as described in Chapter B2.0, to identify any hazardous materials or energy sources that have the potential to initiate an accident that could require further review or analysis. This process resulted in the selection of six candidate accidents for more comprehensive analysis in Section B3.4.2.

B3 3.1 Methodology

The methodology used to identify and evaluate SNF Project facility hazards is described in detail in Section 3.3 of the SNF Project FSAR. This section discusses those areas of the methodology that are specific to the CVDF. The hazard evaluation process identified hazardous conditions, determined causes and preventive and mitigative features, and qualitatively estimated the consequences and frequencies of occurrence (HNF-SD-SNF-HIE-004). The results of the application of this methodology to the CVDF are presented in Section B3.3.2. The hazard analysis was performed in accordance with DOE-STD-3009-94 and implements the requirements of DOE Order 5480.23. Figure B3-1 shows the hazard and accident methodology used for the CVDF.

B3 3.1.1 Hazard Identification During normal operations, small amounts of airborne particulate from the MCO are expected to be lifted off the surface of the SNF as the process system sweeps the MCO with helium. This particulate is filtered at the MCO exit port before entering the process system. In addition, the process system flow is directed through a two-stage, high-efficiency particulate air (HEPA) filter before exiting the stack. No adverse consequences for the public or the work force or contamination of the environment are expected under these normal conditions (normal operating releases are discussed in more detail in Chapter B9.0).

The CVDF hazard analysis covers activities beginning with the approach of the cask-MCO transporter to the facility, anticipated normal facility operations, the MCO drying and proof processes, preparing the cask-MCO and the trailer for shipping, and ending with the transporter leaving the facility. Hazard identification for the CVDF was based on examination of the six major building areas in the facility:

1. Administrative area (AA)
2. Transfer corridor and mechanical equipment room (TC)
3. Process bays 2 through 5 (PB)
4. Process bay 1, an unused bay (SB)
The CVDF processes and activities that can take place within each area were identified, and the hazards were identified by form (e.g., electrical, thermal, friction), type (e.g., motors, power tools, winng), and building location. A standardized hazardous material/energy source checklist was used to group potentially hazardous materials and energy sources in each of the six major facility areas. The two letter abbreviations shown for the six areas are a portion of the unique identification designator used for tracking in the analysis process. If a hazard affected more than one area, it was evaluated in all facility areas where material at risk (MAR) was present. Figure B3-2 shows the standardized hazardous material/energy checklist for the CVDF hazards identification.

B3 3 1 2 Hazard Evaluation The identified hazards were evaluated to determine the causes of the hazard, potential accidents that could result from the presence of each hazard and consequences to the public offsite, collocated and facility workers onsite the environment, or the CVDF. Safety features, segregated into preventive and mitigative features, were identified for each hazard based on the ability of the feature to prevent or mitigate the consequences. Qualitative estimates of the frequency and consequence of the hazardous condition were assigned. See Section 3 3 1 2 of the SNF Project FSAR (HNF-3553) for the criteria used in assigning the consequence and frequency categories.

B3 3 2 Hazard Analysis Results

The hazard analysis process is documented in HNF-SD-SNF-HIE-004. The results of that process in their order of progression, are as follows:

- A series of checklist-style tables describing hazardous materials and energy sources organized by major facility area — These tables were used to develop the hazard analysis accident scenarios.

- A series of tables describing the standard industrial hazards considered, organized by major facility area — These events were judged to have no contribution to uncontrolled radiological and/or hazardous materials releases and were not considered in the selection of DBAs, safety-class or safety-significant features, or TSRs. They were among the hazards considered and, therefore, are included for completeness.

- A series of tables describing potential hazard scenarios, organized by major facility area — These tables included hazardous energy sources and materials, hazardous conditions, causes and initiators, potential accidents, qualitative determinations of event frequencies and consequences, safety features for prevention and/or mitigation of the consequences, and defense-in-depth or worker safety features.
A table, organized by major facility area, assigning risk bins to causes associated with significant consequences to offsite and onsite receptors — Consistent with DOE-STD-3009-94, the events located in risk bins representing "situations of concern," or "situations of major concern," were evaluated as candidate DBAs.

- A final list of candidate DBAs, sorted by risk ranking and energy change or release — This list formed the basis for selecting the DBAs presented in Section B3 4 2.

DBA selection is addressed in Section B3 3 2 3 5 and its accompanying Table B3-4. In terms of the risk binning process, the accidents chosen from the hazard analysis for further analysis as DBAs were all events identified in consequence categories S3, indicating significant effects to offsite receptors, and S2, indicating significant effects to onsite receptors.

The hazard analysis (HNF-SD-SNF-HIE-004) has been reconciled with the up-to-date CVDF facility design described in Chapters B2 0, B4 0, and B5 0 and the accident analyses in this chapter.

**B3 3 2 1 Hazard Identification**

The final CVDF hazard analysis tables are shown in HNF-SD-SNF-HIE-004. A checklist designator (from the hazard analysis tables) associated with each hazard contains a two-letter facility location designation, a letter corresponding to the general type of hazardous energy or material source, and a final number (sometimes with an associated letter) that identifies the specific source of the hazardous energy or material. The main inventory of hazardous material in the CVDF is the radionuclide content of the MCOs. The toxicological hazards of the radionuclide inventory were reviewed. As described in Section 3 4 1 1 of the SNF Project FSAR, the radiological guidelines were found to be more limiting than the toxicological guidelines for the release of the SNF particulate in the MCO. Other hazardous material identified by the hazard identification process includes pyrophoric metals and hydrides, oxidizers, hydrogen, diesel fuel, and other flammable or combustible materials. Table B3-1 provides a summary that identifies hazards by form, type, location, and total quantity.

A specific and comprehensive analysis of all fire hazards associated with the CVDF has been completed (SNF-4268 and SNF-4942) to augment the standard hazard analysis.

The CVDF does not have an operating history, so major hazards resulting from facility operation cannot be identified or summarized as suggested by DOE-STD-3009-94. However, as described in Section 3 3 2 1 of the SNF Project FSAR, the CVDF spent fuel drying process is similar to the vacuum drying process used by independent spent fuel storage installations issued licenses under 10 CFR 72. Major hazards from similar independent spent fuel storage installations were considered when performing the hazard analysis for the CVDF. These hazards include failure of drying processes, defects in cask integrity, and the spread of external contamination (see SNF Project FSAR, Section 3 3 2 1).
<table>
<thead>
<tr>
<th>Field name or location</th>
<th>MAR-subject</th>
<th>MAR-description</th>
<th>MAR-classification</th>
<th>Capacity</th>
<th>Material type</th>
<th>Physical form</th>
<th>Volume or activity</th>
<th>Transient</th>
<th>Quantity</th>
<th>Comments</th>
</tr>
</thead>
<tbody>
<tr>
<td>Process bay</td>
<td>MCO in cask</td>
<td>SNF and particulate matter in MCO (including contaminated water)</td>
<td>SNF in MCO</td>
<td>1,000 L per MCO consisting of five to six baskets, SNF water</td>
<td>Spent fuel from N Reactor (upper bound)</td>
<td>Solid consisting of fuel and particulate corrosion products</td>
<td>Yes, particulate is transient</td>
<td>Upper bound mass at 6.8 MTU per MCO</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Hydrogen gas</td>
<td>Combustible gas</td>
<td>Approximately 1 m² of hydrogen gas per MCO generated over entire CVDF process at pressures from less than 12 torr to 4 lb/in² gauge (normal)</td>
<td>Hydrogen gas</td>
<td></td>
<td></td>
<td>Approximately 1 m² of hydrogen gas per MCO</td>
<td>Yes</td>
<td>Approximately 1 m² hydrogen gas per MCO</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Truck fuel tank</td>
<td>Diesel fuel</td>
<td>Diesel fuel, 80 gal</td>
<td>Fuel</td>
<td>Liquid</td>
<td>Up to 80 gal</td>
<td>Yes</td>
<td>Two tanks</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>SCHe system</td>
<td>Helium gas cylinders</td>
<td>Helium, --240 ft³ per cylinder 4 cylinders</td>
<td>Helium</td>
<td>Gas</td>
<td>--960 ft³ total ( \approx 2,400 ) lb/in² gauge per cylinder</td>
<td>No</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Transfer conduit/ mechanical equipment room</td>
<td>Particulate on HVAC filters (general and local)</td>
<td>Fines divided radioactive particulate from MCOs</td>
<td>Radioactive particulate matter</td>
<td>No more than 94 g</td>
<td>NA</td>
<td>Solid</td>
<td>Up to 94 g</td>
<td>Yes, particulate is transient</td>
<td>94 g</td>
<td></td>
</tr>
<tr>
<td>Process water conditioning room</td>
<td>Process water conditioning system components and piping</td>
<td>SNF and particulate retained from MCO</td>
<td>Radioactive particulate matter</td>
<td>Safety base (housings) of no more than 1.5 kg</td>
<td>Spent fuel from N Reactor</td>
<td>Solid consisting of particulate corrosion products</td>
<td>0.0013 MTU at 26,200 Cs/MTU</td>
<td>Yes, particulate is transient</td>
<td>Upper bound is 1.5 kg</td>
<td></td>
</tr>
<tr>
<td>Outside area</td>
<td>Inert gas storage</td>
<td>Helium used to inert MOB and overpack storage tubes</td>
<td>Helium</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Standby power diesel supply</td>
<td>Diesel fuel</td>
<td>Diesel fuel, 550 gal</td>
<td>Fuel</td>
<td>Liquid</td>
<td>550 gal</td>
<td>No</td>
<td>Day tank and supply tank</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

CSB = Canister Storage Building.
CVDF = Cold Vacuum Drying Facility
HVAC = heating, ventilation, and air conditioning (filter)
IMO = in-core exchange module
MAR = material at risk
MCO = multi-canister overpack
MTU = metric ton of uranium
NA = not applicable
SNF = spent nuclear fuel.
B3 3 2 2 Hazard Classification  A final hazards categorization of the CVDF has been performed based on the final hazard analysis (HNF-SD-SNF-HIE-004) and the accident analysis documentation (SNF-2770) for the facility  Consistent with DOE-STD-1027-92, the final categorization was based on MAR chosen to bound the MAR from the worst-case DBA in Section B3 4 2 These CVDF material quantities have been compared against the DOE-STD-1027-92 threshold quantities  The CVDF final hazard categorization found the CVDF to be a hazard category 2 nuclear facility

The established safety basis radiological nuclide inventory (fuel activity) per metric ton of uranium (HNF-SD-SNF-SARR-005) was used to estimate the material quantities available for release by multiplying the MAR and the specific nuclide inventories (fuel activity) per metric ton of uranium (see Table B3-2)  For any DBA described in this section, the maximum MAR is less than 200 kg UO₂ dispensible particulate (Section B3 4 2 5)  This quantity is compared against the category 2 threshold quantities from DOE-STD-1027-92, Table A 1 (see Table B3-2)  The ratios of the constituents of the MAR to the category 2 threshold quantities were calculated and summed to compare with the summation of the radionuclide ratios for category 2 threshold criteria  Since one radionuclide, ²⁴¹Am, is greater than the corresponding hazard category 2 threshold quantity, and since the sum of ratios is greater than one, the CVDF remains a category 2 nuclear facility as identified during the preliminary hazard classification

The hazard classification process does not rely on segmentation of the radionuclide inventory (HNF-SD-SNF-DRD-002)  The final hazard analysis does not identify any toxicological consequences or any intentional chemical process systems, except the ion-exchange resin system  Analysis shows that the radiological guidelines are more limiting than the toxicological guidelines, as discussed in Section 3 4 1 1 of the SNF Project FSAR  The categorization level would not change if the quantity of MAR were to increase because the CVDF cannot be a category 1 facility unless designated category 1 by order of the responsible DOE program senior official (DOE-STD-1027-92)

B3 3 2 3 Hazard Evaluation  The final CVDF hazard analysis identified hazards associated with actual processes used in the CVDF (HNF-SD-SNF-HIE-004)  Standard industrial hazards were identified (HNF-SD-SNF-HIE-004, Tables 9 through 14) and removed from the list of facility hazards used to identify DBAs  The results of the hazard analysis identified hazard scenarios for each major area in the CVDF  These scenarios were used to define the CVDF DBAs selected for further analysis in Section B3 4 2

Evaluation of nearby external human-generated activities that may represent a threat to the facility is required by DOE-STD-3009-94 and is an additional NRC equivalency requirement specified in HNF-SD-SNF-DB-003  The external hazards from the human-generated activity identified in Section B1 6 involve only those from aircraft activity  Section B1 6 identifies 10 active airports within a 24-mile radius of the CVDF (Beary 1997) and defines the overall frequency of aircraft impact from all sources to be 4.6 x 10⁻⁷ per year  The total impact frequency is less than 10⁻⁶ per year and therefore the safety risk is below the level of concern  As stated in Section B1 6, nearby truck, rail, and barge traffic does not exceed the guidelines for shipments of hazardous materials that could present a risk to the CVDF
### Table B3-2  Radionuclide Inventory for the Cold Vacuum Drying Facility Material at Risk Compared with the Category 2 Threshold Quantities  (3 sheets)

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Fuel activity (Ci/MTU)</th>
<th>Material at risk* (200 kg [0.2 MTU] Cl)</th>
<th>Category 2 threshold quantitiesb</th>
<th>Ratio of material to category 2 threshold quantity</th>
</tr>
</thead>
<tbody>
<tr>
<td>H3</td>
<td>2.61 E+01</td>
<td>5.22 E+00</td>
<td>3.00 E+05</td>
<td>1.74 E-05</td>
</tr>
<tr>
<td>C-14</td>
<td>5.53 E+01</td>
<td>1.11 E+01</td>
<td>1.40 E+06</td>
<td>7.90 E-08</td>
</tr>
<tr>
<td>Fe-55</td>
<td>5.41 E+01</td>
<td>1.08 E+01</td>
<td>1.10 E+07</td>
<td>9.84 E-09</td>
</tr>
<tr>
<td>Co-60</td>
<td>2.09 E+00</td>
<td>4.18 E+01</td>
<td>1.90 E+05</td>
<td>2.20 E-06</td>
</tr>
<tr>
<td>Ni-59</td>
<td>3.18 E-02</td>
<td>6.36 E-03</td>
<td>4.30 E+05</td>
<td>1.48 E-08</td>
</tr>
<tr>
<td>Ni-63</td>
<td>3.47 E+00</td>
<td>6.94 E+01</td>
<td>4.50 E+06</td>
<td>1.54 E-07</td>
</tr>
<tr>
<td>Se 79</td>
<td>6.54 E+02</td>
<td>1.31 E+02</td>
<td>4.30 E+05</td>
<td>3.04 E-08</td>
</tr>
<tr>
<td>Kr 85</td>
<td>3.70 E+02</td>
<td>7.40 E+01</td>
<td>2.80 E+07</td>
<td>2.64 E-06</td>
</tr>
<tr>
<td>Sr 90</td>
<td>6.93 E+03</td>
<td>1.39 E+03</td>
<td>2.20 E+04</td>
<td>6.30 E-02</td>
</tr>
<tr>
<td>Y 90</td>
<td>6.93 E+03</td>
<td>1.39 E+03</td>
<td>4.30 E+05</td>
<td>3.22 E-03</td>
</tr>
<tr>
<td>Zr 93</td>
<td>2.95 E+01</td>
<td>5.90 E-02</td>
<td>8.90 E+04</td>
<td>6.63 E-07</td>
</tr>
<tr>
<td>Nb 93m</td>
<td>1.93 E+01</td>
<td>3.86 E+02</td>
<td>4.30 E+05</td>
<td>8.98 E-08</td>
</tr>
<tr>
<td>Tc 99</td>
<td>2.19 E+00</td>
<td>4.38 E+01</td>
<td>3.80 E+06</td>
<td>1.15 E-07</td>
</tr>
<tr>
<td>Ru 106</td>
<td>2.56 E+02</td>
<td>5.12 E+03</td>
<td>6.50 E+03</td>
<td>7.88 E-07</td>
</tr>
<tr>
<td>Rh 106</td>
<td>2.56 E+02</td>
<td>5.12 E+03</td>
<td>4.30 E+05</td>
<td>1.19 E-08</td>
</tr>
<tr>
<td>Pd 107</td>
<td>1.56 E+02</td>
<td>3.12 E+03</td>
<td>4.30 E+05</td>
<td>7.26 E-09</td>
</tr>
<tr>
<td>Ag 110</td>
<td>7.17 E+00</td>
<td>1.43 E+10</td>
<td>4.30 E+05</td>
<td>3.33 E 16</td>
</tr>
<tr>
<td>Ag 110m</td>
<td>5.39 E-08</td>
<td>1.08 E-08</td>
<td>5.30 E+05</td>
<td>2.03 E 14</td>
</tr>
<tr>
<td>Cd 113m</td>
<td>2.78 E+00</td>
<td>5.56 E-01</td>
<td>4.30 E+05</td>
<td>1.29 E-06</td>
</tr>
<tr>
<td>In 113m</td>
<td>1.36 E+19</td>
<td>2.72 E+20</td>
<td>4.30 E+05</td>
<td>6.33 E 26</td>
</tr>
<tr>
<td>Sn 113</td>
<td>1.36 E+19</td>
<td>2.72 E+20</td>
<td>3.20 E+06</td>
<td>8.50 E 27</td>
</tr>
<tr>
<td>Sn 119m</td>
<td>6.14 E-08</td>
<td>1.23 E+08</td>
<td>4.30 E+05</td>
<td>2.86 E 14</td>
</tr>
<tr>
<td>Sn 121m</td>
<td>6.27 E-02</td>
<td>1.25 E+02</td>
<td>4.30 E+05</td>
<td>2.92 E 08</td>
</tr>
<tr>
<td>Sn 123</td>
<td>1.72 E+16</td>
<td>3.44 E+17</td>
<td>4.30 E+05</td>
<td>8.00 E 23</td>
</tr>
<tr>
<td>Sn 126</td>
<td>1.29 E+01</td>
<td>2.58 E+02</td>
<td>3.30 E+05</td>
<td>7.82 E-08</td>
</tr>
<tr>
<td>Sb 125</td>
<td>0.00 E+00</td>
<td>0.00 E+00</td>
<td>4.30 E+05</td>
<td>0.00 E+00</td>
</tr>
<tr>
<td>Sb 126</td>
<td>1.81 E-02</td>
<td>3.62 E-03</td>
<td>2.50 E+06</td>
<td>1.45 E-09</td>
</tr>
<tr>
<td>Sb 126m</td>
<td>1.29 E-01</td>
<td>2.58 E-02</td>
<td>4.30 E+05</td>
<td>6.00 E-08</td>
</tr>
</tbody>
</table>
Table B3-2 Radionuclide Inventory for the Cold Vacuum Drying Facility Material at Risk Compared with the Category 2 Threshold Quantities (3 sheets)

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Fuel activity (Ci/MTU)</th>
<th>Material at risk* (200 kg [0.2 MTU] Ci)</th>
<th>Category 2 threshold quantities*</th>
<th>Ratio of material to category 2 threshold quantity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Te 123m</td>
<td>1.50 E+21</td>
<td>3.00 E+22</td>
<td>4.30 E+05</td>
<td>6.98 E+28</td>
</tr>
<tr>
<td>Te 125m</td>
<td>0.00 E+00</td>
<td>0.00 E+00</td>
<td>4.30 E+05</td>
<td>0.00 E+00</td>
</tr>
<tr>
<td>Te 127</td>
<td>2.12 E+19</td>
<td>4.24 E+20</td>
<td>4.30 E+05</td>
<td>9.86 E+26</td>
</tr>
<tr>
<td>Te 127m</td>
<td>2.16 E+19</td>
<td>4.32 E+20</td>
<td>1.50 E+05</td>
<td>2.88 E+25</td>
</tr>
<tr>
<td>I 129</td>
<td>5.16 E-03</td>
<td>1.03 E-03</td>
<td>4.30 E+05</td>
<td>2.40 E-09</td>
</tr>
<tr>
<td>Cs 134</td>
<td>6.47 E+00</td>
<td>1.29 E+00</td>
<td>6.00 E+04</td>
<td>2.16 E-05</td>
</tr>
<tr>
<td>Cs 135</td>
<td>6.04 E-02</td>
<td>1.21 E-02</td>
<td>4.30 E+05</td>
<td>2.81 E-08</td>
</tr>
<tr>
<td>Cs 137</td>
<td>9.66 E+03</td>
<td>1.93 E+03</td>
<td>8.90 E+04</td>
<td>2.17 E-02</td>
</tr>
<tr>
<td>Ba 137m</td>
<td>9.14 E+03</td>
<td>1.83 E+03</td>
<td>4.30 E+05</td>
<td>4.25 E-03</td>
</tr>
<tr>
<td>Ce 144</td>
<td>7.91 E-04</td>
<td>1.58 E-04</td>
<td>8.20 E+04</td>
<td>1.93 E-09</td>
</tr>
<tr>
<td>Pr 144</td>
<td>7.82 E-04</td>
<td>1.56 E-04</td>
<td>4.30 E+05</td>
<td>3.64 E-10</td>
</tr>
<tr>
<td>Pr 144m</td>
<td>9.48 E-06</td>
<td>1.90 E-06</td>
<td>4.30 E+05</td>
<td>4.41 E-12</td>
</tr>
<tr>
<td>Pm 147</td>
<td>1.09 E+02</td>
<td>2.18 E+01</td>
<td>8.40 E+05</td>
<td>2.60 E-05</td>
</tr>
<tr>
<td>Sm 151</td>
<td>1.02 E+02</td>
<td>2.04 E+01</td>
<td>9.90 E+05</td>
<td>2.06 E-05</td>
</tr>
<tr>
<td>Eu 152</td>
<td>8.45 E-01</td>
<td>1.69 E-01</td>
<td>1.30 E+05</td>
<td>1.30 E-06</td>
</tr>
<tr>
<td>Eu 154</td>
<td>1.13 E+02</td>
<td>2.26 E+01</td>
<td>1.10 E+05</td>
<td>2.05 E-04</td>
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<tr>
<td>Eu 155</td>
<td>1.06 E+01</td>
<td>2.12 E+00</td>
<td>7.30 E+05</td>
<td>2.90 E-06</td>
</tr>
<tr>
<td>Gd 153</td>
<td>5.19 E+19</td>
<td>1.04 E+19</td>
<td>1.40 E+06</td>
<td>7.41 E-26</td>
</tr>
</tbody>
</table>

**Actinides**

| U 234         | 3.84 E 01              | 7.68 E 02                              | 2.20 E+02                      | 3.49 E-04                                     |
| U 235         | 1.27 E 02              | 2.54 E-03                              | 2.40 E+02                      | 1.06 E-05                                     |
| U 236         | 7.16 E-02              | 1.43 E-02                              | 5.50 E+01                      | 5.97 E-05                                     |
| U 238         | 3.31 E 01              | 6.62 E-02                              | 2.40 E+02                      | 2.76 E-04                                     |
| Np 237        | 4.66 E 02              | 9.32 E-03                              | 5.80 E+01                      | 1.61 E-04                                     |
| Pu 238        | 1.33 E+02              | 2.66 E+01                              | 6.20 E+01                      | 4.29 E-01                                     |
| Pu 239        | 1.73 E+02              | 3.46 E+01                              | 5.60 E+01                      | 6.18 E 01                                     |
| Pu 240        | 1.37 E+02              | 2.74 E+01                              | 5.50 E+01                      | 4.89 E 01                                     |
| Pu 241        | 6.82 E+03              | 1.36 E+03                              | 2.90 E+03                      | 4.70 E 01                                     |
| Pu 242        | 8.71 E-02              | 1.74 E-02                              | 5.50 E+01                      | 3.11 E-04                                     |
**Table B3-2** Radionuclide Inventory for the Cold Vacuum Drying Facility Material at Risk Compared with the Category 2 Threshold Quantities (3 sheets)

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Fuel activity (Ci/MTU)</th>
<th>Material at risk* (200 kg [0.2 MTU] Cl)</th>
<th>Category 2 threshold quantities*</th>
<th>Ratio of material to category 2 threshold quantity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Am 241</td>
<td>4.34 E+02</td>
<td>8.68 E+01</td>
<td>5.50 E+01</td>
<td>1.58 E+00</td>
</tr>
<tr>
<td>Am 242</td>
<td>3.71 E-01</td>
<td>7.42 E-02</td>
<td>4.30 E+05</td>
<td>1.73 E-07</td>
</tr>
<tr>
<td>Am 242m</td>
<td>3.72 E-01</td>
<td>7.44 E-02</td>
<td>5.60 E+01</td>
<td>1.33 E-03</td>
</tr>
<tr>
<td>Am-243</td>
<td>2.78 E-01</td>
<td>5.56 E-02</td>
<td>5.50 E+01</td>
<td>1.01 E-03</td>
</tr>
<tr>
<td>Cm 242</td>
<td>3.08 E-01</td>
<td>6.16 E-02</td>
<td>1.70 E+03</td>
<td>3.62 E-05</td>
</tr>
<tr>
<td>Cm 244</td>
<td>4.47 E+00</td>
<td>8.94 E+01</td>
<td>5.50 E+01</td>
<td>1.63 E-02</td>
</tr>
</tbody>
</table>

Sum of category 2 ratios (inventory material/threshold quantities) | 3.70 E+00

Note: Radionuclide values are from HNF SD SNF TI-015 1998 *Spent Nuclear Fuel Project Technical Databook* Rev 6 Fluor Daniel Hanford Incorporated Richland Washington

*a The material at risk used to calculate the final Cold Vacuum Drying Facility hazard category bounds the material at risk in the worst-case unmitigated design basis accident

*b Category threshold values are from Table A.1 of DOE STD 1027.92 Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23 Nuclear Safety Analysis Reports

MTU = metric ton of uranium
Threats from natural phenomena and nearby facilities are evaluated in HNF-SD-SNF-HIE-004 and Chapter B3. As described in detail in Chapter B2, the design criteria for natural phenomena were defined based on applicable DOE orders and NRC equivalency requirements, as documented in WHC-SD-SNF-DB-010. These design criteria have been incorporated into the design and engineering of SSCs, as described in Chapter B4. Therefore, credible natural phenomena hazards are addressed in the design of the facility.

Threats from nearby facilities are identified and evaluated in Section B1.7 based on applicable DOE orders, regulations, and NRC equivalency requirements. Potential hazards to the CVDF from offsite hazardous operations or facilities are examined under three general classifications:

- Nonreactor nuclear and nonnuclear industrial facilities within 5 mi of the CVDF
- Nuclear reactors within an 50-mi radius of the CVDF
- Military activities

No significant risks are identified that could impact CVDF operation.

The CVDF fire hazard analysis (SNF-4268) and associated implementation plan (SNF-4942) identified the fire hazards, fire loading criteria, and appropriate requirements for addressing fire hazards during CVDF operation, based on applicable DOE orders, regulations, and NRC equivalency requirements. The fire hazard analysis evaluated several fires that involved either the tractor and trailer, the trailer only, the MCO, or various safety equipment. The fire included the combustible loading of the unit, including the tires, fuel, tractor cab, and special work permit clothing. Fire protection features (e.g., fire detection equipment and sprinklers) for the CVDF are described in Section B11.4. The DBA analyses do not develop separate fire scenarios but instead rely on the scenarios developed in the fire hazard analysis (SNF-4268) and implementation plan (SNF-4942). The controls on combustibles are identified in the pertinent DBA tables in Section B3.4.2, and in Chapter B5.

**B3.3.2.3.1 Planned Design and Operational Safety Improvements** This section is used to discuss commitments for planned major design or operational improvements for the facility that are not yet implemented. Because the facility design and accident analysis have been developed in parallel, the opportunity to provide feedback for design consideration has been available. As the hazard evaluation and specific accident analyses have progressed, cost-effective modifications that improve safety have been incorporated into the design. Therefore, no plans for major design or operational improvements are needed as a result of the hazard evaluation.

**B3.3.2.3.2 Defense in Depth** A summary of fundamental points relevant to the concept of defense in depth is described in Section 3.3.2.3.2 of the SNF Project FSAR.

**Features Chosen to Provide Defense in Depth for the Cold Vacuum Drying Facility** Defense-in-depth features for the CVDF were selected based on a relative ranking of the hazards from the hazard identification process, followed by a selection of safety-class and safety-significant features and TSRs for the DBAs, which are described in Section B3.4.2.
Preventive and mitigative features identified in the hazard analysis (HNF-SD-SNF-HIE-004), but not identified in the accident analyses as safety class, safety significant, or TSRs, are identified as additional defense-in-depth features. The defense-in-depth features are presented in the tables that accompany each DBA in Section B3.4.2. Administrative features identified in these tables are in addition to those already identified in the programmatic chapters (e.g., Chapters B7.0, B8.0, and B11.0).

The initial layer of defense in depth at the CVDF is facility design. All SSCs are designed in accordance with applicable codes and standards with a high degree of reliability, and the design encompasses human factors considerations to ensure that operations can be conducted safely. Other defense-in-depth features for preventing and mitigating hazards and accidents, and for preventing or mitigating radiological consequences to the facility or collocated workers, have also been identified. Recovery actions after a safe and stable state has been reached, if necessary, will be based on analysis and a recovery plan specific to the event developed by an appropriate qualified management team. Chapter B17.0 describes the unreviewed safety question process that must be used for recovery actions, if those actions are not within the bounds of the authorization basis provided by this FSAR.

The MCO internal HEPA filter connected to the filtered process exit port provides a measure of defense in depth in nearly all accidents involving the MCO. However, because the filter is not testable when in place, no credit is taken for it in the DBA analyses. In practical terms, the filter provides for contamination control of process lines and helps to keep worker exposures as low as reasonably achievable (ALARA). The internal HEPA filter is metallic and will continue to filter if wetted, and it will not plug under any identified condition (SNF-4636).

Operator training, approved operating and maintenance procedures (as indicated by vendor-supplied information), and routine maintenance provide a measure of protection against nearly all operational hazards. To avoid unnecessary repetition, these features are not stated in the DBAs as providing defense in depth.

Safety-Significant Structures, Systems, and Components Safety-significant SSCs are predominantly required to prevent or mitigate consequences of postulated accident events to the collocated onsite worker. DOE Letter 97-SFD-172 (Sellers 1997) provides a correlation of the evaluation guidelines to identification of the safety-significant SSCs. In addition, DOE-STD-3009-94 suggests that SSCs be designated as safety significant if they play a key role in defense in depth (or worker safety). The severity of the event being prevented or mitigated and the number of barriers present are provided in DOE-STD-3009-94 as guidance for the identification of defense-in-depth safety-significant SSCs. No safety-significant defense-in-depth SSCs have been identified for the CVDF outside of those features already identified as safety significant in the DBA.

Technical Safety Requirements TSRs were identified for postulated accident events that could challenge accident consequence release limits and evaluation guidelines for the offsite public and collocated onsite worker. TSRs are identified in the individual DBA sections and further explained in Chapter B5.0. In addition, criticality prevention features are controlled by TSRs, as
identified in Chapter 50 of the SNF Project FSAR and Chapters B50 and B60. Only one feature has been identified that requires TSR coverage based on defense in depth considerations; a TSR is required to perform pressure tests on the MCO to minimize hydrogen generation within the cask during transport to the CSB.

**B3.3.2.3 Worker Safety**  Worker safety for the CVDF is ensured by a combination of design features that reduce exposure to radioactive, toxic, and industrial hazards and by institutional practices that, in total, protect workers from these hazards. Protecting the facility worker from the standard industrial hazards identified for the CVDF is achieved through adherence to the institutional safety programs described in Chapters 70, 80, 90, 100, 150, and 170 of the SNF Project FSAR and documented in lower-tier documents such as health and safety plans and job hazards analyses. Such industrial hazards do not require specific safety-significant SSCs or TSR-level administrative controls. Therefore, in accordance with the guidance of DOE-STD-3009-94, the remainder of this section deals with protecting workers from those hazards of facility operation that are exclusive of standard industrial hazards.

The final CVDF hazard analysis provides an overview of the major features protecting facility workers at the CVDF (see HNF-SD-SNF-HIE-004). Worker safety features are an integral part of facility design and operation. The major features of worker protection are identified in Table B3-3 (based on the hazards analysis) and are categorized by hazard. The features presented in Table B3-3 are in addition to those identified as safety-class or safety-significant features in the DBA sections. The hazard energy source or material and the hazardous condition are identified along with the features protecting the worker, including passive, active, and administrative protective features. No safety-significant SSCs or TSRs have been identified for the CVDF based solely on worker safety considerations. The controls identified in the analysis of the DBAs, in conjunction with the safety features identified in the hazard analysis and the institutional programs, are adequate to ensure worker safety. In one case, a safety-class SSC provides a safety-significant function that is not identified in the DBA analysis. The cask and MCO shield plug are credited as safety-class SSCs for establishing initial conditions and allowing other safety-class SSCs to properly function. The shielding function, however, is not identified in the DBA analysis, but is safety significant based on the potential for high dose rates that would be associated with an unshielded MCO.

In general, items included in Table B3-3 exclude standard industrial hazards, hazards that are beyond extremely unlikely and hazards that have no safety consequences. In addition, the features listed in the table exclude features and controls listed in the controls tables for the DBAs and include only those features from the hazard analysis pertaining to worker safety.

**B3.3.2.4 Environmental Protection**  The hazard to the environment from CVDF operations involves the potential release of radioactive contaminants. The predominant release pathway for these contaminants is via the air to the boundaries and receptors discussed in Section B1.3.13. While accidents involving liquid releases have been identified, the quantities and concentrations of the liquids and contaminants are not large enough to result in a significant release to the environment as evaluated in HNF-SD-SNF-HIE-004. Where liquid releases are possible, they are contained by design of the facility (e.g., the sump on the process water conditioning [PWC] skid).
### Table B3-3 Hazards and Safety Features Related to Worker Safety (8 sheets)

<table>
<thead>
<tr>
<th>Hazard energy source/material</th>
<th>Checklist designator*</th>
<th>Hazardous condition</th>
<th>Worker safety features</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kinetic energy linear other</td>
<td>AA F 06 TC F 07 PB F 07 SB F 07 PW F 07</td>
<td>Injury due to compressed gas bottle failure</td>
<td>Gas bottles handled in accordance with approved procedure, Bottle design precludes missile hazard</td>
</tr>
<tr>
<td>Thermal electrical equipment heaters</td>
<td>PB B 02a PB B 02b PB B 03a</td>
<td>Injury due to thermal runaway overpressurization and possible internal/external hydrogen explosion resulting in radioactive particulate release</td>
<td>HVAC sweeps H₂ away from operators, Continuous air monitor local area alarm, Response procedure evacuate, Radiological training</td>
</tr>
<tr>
<td>Thermal MCO contents</td>
<td>PB B 13a PB B 13b PB B 13c PB B 13d</td>
<td>Injury due to thermal runaway overpressurization and possible internal/external hydrogen explosion resulting in radioactive particulate release</td>
<td>HVAC sweeps H₂ away from operators, Continuous air monitor local area alarm, Response procedure evacuate, Radiological training</td>
</tr>
<tr>
<td>Thermal MCO contents</td>
<td>PB B 13g</td>
<td>Injury due to external hydrogen explosion resulting in radioactive particulate release</td>
<td>HVAC sweeps H₂ away from operators, Continuous air monitor local area alarm, Response procedure evacuate, Radiological training</td>
</tr>
<tr>
<td>Linear kinetic transportation cask trailer</td>
<td>PB F 01b PB F 01c</td>
<td>Injury due to trailer impact with process equipment and subsequent radioactive particulate release</td>
<td>Physical barrier or indicator near the process bay doors and the back bay wall to prevent over travel, The mezzanine supports protect the process skid from damage, Vehicles are operated by qualified personnel and enter and exit the bays according to approved procedures, The floor is painted to aid in guiding the transporter correctly, Continuous air monitor local area alarm, Response procedure evacuate, Radiological training</td>
</tr>
<tr>
<td>Linear kinetic forklifts dollies carts</td>
<td>PB F 02a</td>
<td>Injury due to forklift impact with process equipment and subsequent radioactive particulate release</td>
<td>Continuous air monitor local area alarm, Response procedure evacuate, Radiological training, Vehicles are operated by qualified personnel and enter and exit the bays according to approved procedures</td>
</tr>
<tr>
<td>Hazard energy source/material</td>
<td>Checklist designator*</td>
<td>Hazardous condition</td>
<td>Worker safety features</td>
</tr>
<tr>
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</tr>
</tbody>
</table>
| Linear kinetic crane loads   | PB F 05PB G 03a       | Injury due to crane loaded with cask lid or process hood resulting in thermal runaway and internal/external hydrogen explosion resulting in radioactive particulate release | Crane is operated by qualified personnel following approved procedures  
Crane maintenance conducted according to approved procedures  
Crane movement restricted by design  
All lifts by crane passing over safety class equipment follow DOE guidelines for such lifts  
HVAC sweeps H₂ away from operators  
Continuous air monitor local area alarm  
Response procedure evacuate  
Radiological training |
| Linear kinetic cars trucks buses forklifts dollies carts | SB F 01b SB F 02b | Injury due to vehicle striking process lines resulting in internal/external hydrogen explosion and subsequent radioactive particulate release | Vehicles are operated by qualified personnel and enter and exit the bays according to approved procedures  
Response procedure evacuate  
Radiological training |
| Linear kinetic cars trucks buses forklifts dollies carts | SB F 01c SB F 02c | Injury due to vehicle striking MCO drain line resulting in radioactive liquid and particulate release | Vehicles are operated by qualified personnel and enter and exit the bays according to approved procedures  
Response procedure evacuate  
Radiological training |
| Mass gravity height lifts and cranes | PB F 05 PB G 03a | Injury due to crane loaded with cask lid or process hood impacts with process systems and subsequent radioactive liquid and particulate release | Crane maintenance is conducted according to approved procedures  
Crane movement restricted by design  
Crane is operated by qualified personnel following approved procedures  
Continuous air monitor local area alarm  
Response procedure evacuate  
Radiological training  
HVAC sweeps H₂ away from operators  
All lifts by crane passing over safety class equipment follow DOE guidelines for such lifts |

* Checklist designator is a code used for identification.
<table>
<thead>
<tr>
<th>Hazard energy source/material</th>
<th>Checklist designator*</th>
<th>Hazardous condition</th>
<th>Worker safety features</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mass gravity height elevated crane trolley</td>
<td>PB G 03b</td>
<td>Injury due to crane trolley impacts process systems and subsequent radioactive liquid and particulate release</td>
<td>Crane designed with overlift protection. Personnel are trained in the proper use of the crane and the correct procedure for processing and follow approved procedures for same. Crane maintenance is conducted according to approved procedures. Continuous air monitor local area alarm. Response procedure evacuate. Radiological training.</td>
</tr>
<tr>
<td>Mass gravity height elevated cask lid</td>
<td>PB G 03c</td>
<td>Elevated cask lid falling could potentially damage MCO injure personnel</td>
<td>Crane designed with overlift protection. Personnel are trained in the proper use of the crane and the correct procedure for processing and follow approved procedures for same. Crane maintenance is conducted according to approved procedures. Continuous air monitor local area alarm. Response procedure evacuate. Radiological training.</td>
</tr>
<tr>
<td>Pressure volume pressure vessels (MCO and cask)</td>
<td>PB H 06a</td>
<td>Injury due to pressurized release of hydrogen from MCO during lid removal or line hookup and subsequent ignition</td>
<td>Personnel trained in the correct process steps prior to connecting process lines. HVAC sweeps H₂ away from operators. Continuous air monitor local area alarm. Response procedure evacuate. Radiological training.</td>
</tr>
</tbody>
</table>
Table B3-3 Hazards and Safety Features Related to Worker Safety (8 sheets)

<table>
<thead>
<tr>
<th>Hazard energy source/material</th>
<th>Checklist designator*</th>
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<th>Worker safety features</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pressure volume pressure vessels</td>
<td>PB H 06b</td>
<td>Injury due to release of annulus water from between the MCO and the transportation cask resulting in spray or leak of contaminated water</td>
<td>Alarm occurs on seal ring pressure loss through the monitoring and control system</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Operations check for leakage</td>
</tr>
<tr>
<td></td>
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<td></td>
<td>The male end of the connectors which are less likely to have leaks than female connections for the tempered water system connections are on the transportation cask (This connection is to be checked once per year by the transportation system personnel)</td>
</tr>
<tr>
<td></td>
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<td></td>
<td>The cask loading process uses a system to maintain clean water in the cask, minimizing the expected contamination</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Tempered water system connection to cask is double walled</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>Continuous air monitor local area alarm</td>
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<td></td>
<td>Response procedure evacuate</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Radiological training</td>
</tr>
<tr>
<td>Pressure volume pressure vessels</td>
<td>PB H 06c OU P 04</td>
<td>Injury due to pressurized relief from MCO resulting in external hydrogen explosion and pressurized radioactive particulate release</td>
<td>The MCO is vented by means of a rupture disk (credited as a passive design feature)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>HVAC sweeps H₂ away from operators</td>
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<td></td>
<td></td>
<td></td>
<td>Continuous air monitor local area alarm</td>
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<td></td>
<td>Response procedure evacuate</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Radiological training</td>
</tr>
<tr>
<td>Pressure volume pressure vessels</td>
<td>PB H 06d PB H 06e Refer to PB B 03a</td>
<td>Injury due to pressurized relief from process line failure resulting in pressurized radioactive particulate release or pressurized external hydrogen explosion and radioactive particulate release in local exhaust or PWC system</td>
<td>Continuous air monitor local area alarm</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Response procedure evacuate</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Radiological training</td>
</tr>
<tr>
<td>Pressure volume pressure vessels</td>
<td>PB H 06f PB H 06i PB H 06k Refer to PB B 13a</td>
<td>Injury due to thermal runaway or internal/external hydrogen explosion resulting in radioactive particulate release</td>
<td>HVAC sweeps H₂ away from operators</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Continuous air monitor local area alarm</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Response procedure evacuate</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Radiological training</td>
</tr>
</tbody>
</table>
### Table B3-3  Hazards and Safety Features Related to Worker Safety  
(8 sheets)

<table>
<thead>
<tr>
<th>Hazard energy source/material</th>
<th>Checklist designator*</th>
<th>Hazardous condition</th>
<th>Worker safety features</th>
</tr>
</thead>
</table>
| Pressure volume pressure vessels | PB H 06h  
PB H 06l | Injury due to pressurized relief from MCO due to loss of local ventilation resulting in radioactive particulate release to one or more bays or injury due to uncontrolled release of water from MCO resulting in spray or leak of radioactive particulate | Continuous air monitor local area alarm  
Response procedure evacuate  
Radiological training  
Ventilation system design prevents flow of contamination from affected bay to other bays or facility areas |
| Pressure volume pressure vessels | PB H 11c | Injury due to uncontrolled release of water back to MCO (reverse flow) resulting in spray or leak of radioactive particulate | Continuous air monitor local area alarm  
Response procedure evacuate  
Radiological training |
| Pressure volume pressure vessels other | PB H 11a | Injury due to overpressurization of MCO due to high helium system pressure resulting in radioactive particulate release | Continuous air monitor local area alarm  
Response procedure evacuate  
Radiological training  
Pressure relief device on helium trailer |
| Pressure volume other (pressurized process lines) | PB H 11b | Injury due to an external hydrogen explosion resulting in radioactive particulate release | HVAC sweeps H₂ away from operators  
Continuous air monitor local area alarm  
Response procedure evacuate  
Radiological training |
| Pressure volume other (water addition to MCO) | PB H 11d  
PB H 11e  
PB H 11f  
Also PB B 13b | Injury due to internal or external hydrogen explosion thermal runaway or overpressurization resulting in radioactive particulate release | Cask-MCO structural design no inleakage path for tempered water into MCO  
Check valves in lines  
Continuous air monitor local area alarm  
Response procedure evacuate  
Radiological training |
| Pressure volume other (building zoned pressure boundaries) | SB H 11 | Injury due to potential radioactive particulate release from helium overpressurizing PWC drain line | Safety relief valves installed in helium supply |
| Pressure vessels | PW H 06 | Injury due to gaseous and spray release from pressurized components resulting in radioactive particulate release | PWC room designed to contain spills  
HPT survey prior to recovery  
Continuous air monitor local area alarm  
Response procedure evacuate  
Radiological training |
| Pressure | PW H 11 | Injury due to failure of drain line resulting in high pressure helium blowdown | Safety relief valves installed in helium supply |
Table B3-3 Hazards and Safety Features Related to Worker Safety (8 sheets)

<table>
<thead>
<tr>
<th>Hazard energy source/material</th>
<th>Checklist designator*</th>
<th>Hazardous condition</th>
<th>Worker safety features</th>
</tr>
</thead>
<tbody>
<tr>
<td>Explosives/ pyrophorics /Flammable materials hydrogen</td>
<td>PW J 06, PW L 11</td>
<td>Injury due to external hydrogen explosion resulting in radioactive particulate release</td>
<td>PWC drain line and receiver tanks purged with helium prior to draining, HPT survey prior to recovery, Continuous air monitor local area alarm, Response procedure evacuate, Radiological training, Oxygen monitor local area alarm</td>
</tr>
<tr>
<td>Nuclear criticality temporary storage areas filters tanks canals and basins</td>
<td>PW K 02, PW K 04, PW K 15, PW K 08</td>
<td>Injury due to buildup of fissile material resulting in a criticality event</td>
<td>HPT survey prior to recovery, Area radiation monitor local area alarm, Response procedure evacuate, Radiological training</td>
</tr>
<tr>
<td>Flammable / combustible materials rags gasoline lubricating oil coolant oil paint solvent, diesel fuel buildings and contents trailers and contents grease propane alcohol aerosol propellent and external sources</td>
<td>PB L 01, PB L 02, PB L 03, PB L 04, PB L 05, PB L 06, PB L 07, PB L 08, PB L 09, PB L 10, PB L 13, PB L 14, PB L 15, PB L 16, OU P 02a</td>
<td>Injury due to thermal runaway and internal/external hydrogen explosion resulting in radioactive particulate release</td>
<td>Combustible loading controlled to the DBA limits, Fire protection system is present (sprinklers) in bays and adjacent areas, SCIC high bay temperature trip, The transporter cab is removed promptly reducing the potential for a spill due to its limited presence, HVAC sweeps H₂ away from operators, Continuous air monitor local area alarm, Response procedure evacuate, Radiological training, Area around building maintained clear of combustible materials</td>
</tr>
<tr>
<td>Flammable materials hydrogen</td>
<td>PB L 11a, PB L 11b, PB L 11c, PB L 11d, PB L 11e, PB L 11g, Also PB B 13g</td>
<td>Injury due to external hydrogen explosion and pressurized radioactive particulate release</td>
<td>HVAC sweeps H₂ away from operators, Continuous air monitor local area alarm, Response procedure evacuate, Radiological training</td>
</tr>
</tbody>
</table>
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<th>Worker safety features</th>
</tr>
</thead>
<tbody>
<tr>
<td>Flammable materials rags buildings and contents propane aerosol propellant</td>
<td>PW L 02, PW L 08, PW L 14, PW L 16</td>
<td>Injury due to fire failing process equipment resulting in spray release and subsequent radioactive liquid release</td>
<td>Combustible loading controlled to the DBA analysis limits Fire protection system is present (sprinklers) HPT survey prior to entry Continuous air monitor local area alarm Response procedure evacuate Radiological training</td>
</tr>
<tr>
<td>Hazardous materials heavy metals (TRU)</td>
<td>SB M 08, PB M 08</td>
<td>Injury due to TRU in drain line</td>
<td>HPT routine surveys conducted Response procedure evacuate Radiological training</td>
</tr>
<tr>
<td>Hazardous materials asphyxiants</td>
<td>TC M 02</td>
<td>Injury due to helium exposure from helium line break</td>
<td>System relief valves on helium line</td>
</tr>
<tr>
<td>Ionizing radiation sources fissile material</td>
<td>PB N 01</td>
<td>Injury due to MCO drain line failure resulting in liquid and gaseous release of radioactive particulate release</td>
<td>HPT routine surveys conducted Continuous air monitor local area alarm Response procedure evacuate Radiological training</td>
</tr>
<tr>
<td>Ionizing radiation sources radioactive material</td>
<td>PB N 03a, PB N 03b, PB N 03c</td>
<td>Injury due to increased radiation in process lines</td>
<td>HPT routine surveys conducted Response procedure evacuate Radiological training</td>
</tr>
<tr>
<td>Ionizing radiation sources radioactive material</td>
<td>TC N 03</td>
<td>Injury due to HEPA filter failure resulting in radioactive particulate release</td>
<td>Radiological survey program Radiological material loading on HEPA filters limited for early changeout</td>
</tr>
<tr>
<td>Ionizing radiation sources fissile material</td>
<td>PW N 01</td>
<td>Injury due to liquid release with spray resulting in radioactive particulate release</td>
<td>HPT survey prior to recovery Continuous air monitor local area alarm Response procedure evacuate if in room Radiological training PWC room designed to contain spills</td>
</tr>
<tr>
<td>Ionizing radiation sources radioactive material</td>
<td>PW N 03a</td>
<td>Injury due to direct radiation</td>
<td>HPT survey prior to recovery Area radiation monitor local area alarm Response procedure evacuate Radiological training</td>
</tr>
<tr>
<td>Ionizing radiation sources fissile material</td>
<td>SB N 01</td>
<td>Injury due to gaseous release of radioactive particulate</td>
<td>Response procedure evacuate Radiological training</td>
</tr>
</tbody>
</table>
### Table B3-3 Hazards and Safety Features Related to Worker Safety (8 sheets)

<table>
<thead>
<tr>
<th>Hazard energy source/material</th>
<th>Checklist designator*</th>
<th>Hazardous condition</th>
<th>Worker safety features</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ionizing radiation sources radioactive material</td>
<td>SB N 03a</td>
<td>Injury due to damaged drain line resulting in liquid and gaseous release of radioactive particulate</td>
<td>Vehicles are operated by qualified personnel and enter and exit the bays according to approved procedures</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Floor drains and catch tank collect and contain released liquid</td>
</tr>
<tr>
<td></td>
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<td></td>
<td>PWC system interlocks only allow one MCO draining operation to be in effect at a time limiting the available liquid release volumes</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Response procedure evacuate</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Radiological training</td>
</tr>
<tr>
<td>Ionizing radiation sources radioactive material</td>
<td>SB N 03b</td>
<td>Injury due to process water tank room processed water being transferred to tanker truck with potential line break resulting in liquid and gaseous release of radioactive particulate</td>
<td>Transfer conducted by qualified personnel vehicles enter and exit the bays according to approved procedures</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Personnel in room to monitor activity</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>HPT present for surveillance</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Response procedure evacuate</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Radiological training</td>
</tr>
</tbody>
</table>

Note: For those hazards related to the cranes in the process bays there is a TSR restricting crane movement during MCO processing except as part of an approved recovery procedure.

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*Hazard analysis checklist designators are from, HNF SD SNF HIE-004 1999 Cold Vacuum Drying Facility Hazard Analysis Report Rev 4 Fluor Daniel Hanford Incorporated Richland Washington Figure B3.2 shows the hazardous material/energy source checklist that was used to group potentially hazardous materials and energy sources for each of the six major building areas:

AC Administrative area
TC Transfer corridor and mechanical corridor
PB Process bays 2 through 5
SB Process bay 1 an unused bay
PW Process water tank room
OU Outside

DBA = design basis accident
DOE = U.S. Department of Energy
HEPA = high efficiency particulate air (filter)
HPT = health physics technician
HVAC = heating ventilation and air conditioning
MCO = multi canister overpack
PWC = process water conditioning
SCIC = safety-class instrumentation and control
TRU = transuranic
Based on the CVDF design and operating information, no use of toxic chemicals has been identified. The toxicological hazards of the radionuclide inventory have been reviewed. As described in Section 3.4.1.1 of the SNF Project FSAR, the radiological guidelines have been found to be more limiting than the toxicological guidelines (HNF-SD-SNF-TI-059). Potential consequences, including offsite releases, and required prevention and mitigation features are discussed in Section B3.4.2. Implementation of the prevention and mitigation features will prevent releases that could have significant environmental impact. The CVDF features that protect the onsite collocated worker and the offsite public against radiological exposure also serve to prevent and mitigate radiological release to the environment. In addition, sitewide programs for environmental monitoring provide for assessment of the impact of facility releases. Normal CVDF operational activities are expected to have a minor impact on the local and regional environment, as noted in DOE/EIS-0245F, Addendum (Final Environmental Impact Statement) Management of Spent Nuclear Fuel from the K Basins at the Hanford Site, Richland Washington.

B3.3.2.3.5 Accident Selection The methodology for the selection of DBAs is specified in DOE-STD-3009-94. DBAs have been selected to ensure that the range of accident scenarios analyzed represents a complete set of representative and bounding conditions. This selection criterion is a common requirement among the SNF Project facilities and is described in Section 3.3.2.3.5 of the SNF Project FSAR.

The list of six candidate accidents that resulted from the hazards binning process for the CVDF is presented in Table B3-4. The table also contains all accidents identified as category S3 and S2 bounded by each candidate accident. The candidate accidents, and the controls selected for their prevention and mitigation, are described in DBA Sections B3.4.2.1 through B3.4.2.6. The other events listed in Table B3-4, and the controls selected for their prevention and mitigation, are included in the safety feature summary table provided in each DBA section.

From each risk bin, the accident qualitatively determined to be of highest overall risk (a function of frequency and consequence) and highest consequence (regardless of frequency) was selected as a DBA for further evaluation. Currently, a single accident from each bin meets both criteria (being the highest risk as well as highest in consequence), requiring only one accident to be developed per bin. For bins containing multiple accidents of equal risk rank, the DBA was chosen based on both a qualitative evaluation of which accident presented the greatest consequence due to energy present for release and an evaluation of significant local effects (i.e., potentially great impact to facility personnel).

B3.3.3 Abnormal Events for the Cold Vacuum Drying Facility

The purpose of this section is to develop NRC safety equivalency by analyzing “off-normal or abnormal operations” at the CVDF. Abnormal events are expected to occur during the facility lifetime and include non-accident conditions resulting from situations outside of normal operations, where normal operations are defined by process flow diagrams, system design...
## Table B3-4  Binned Listing of Candidate Accidents Sorted by Risk Ranking  (4 sheets)

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Initiator type</th>
<th>Frequency/consequence categories</th>
<th>Hazard analysis checklist designator</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Gaseous releases (Section B3 4 2 1)</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>G 1  Gaseous release due to process line failure or HVAC failure</td>
<td>O</td>
<td>F3/S2</td>
<td>WB B 13a PB H-06g</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>WB F-05 PB H-06h</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>WB H-06d PB N-01</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>WB H-06f SB N-01</td>
</tr>
<tr>
<td>G 2  Gaseous release due to delays in shipping from the CVDF</td>
<td>O</td>
<td>F2/S2</td>
<td>WB H-06i</td>
</tr>
<tr>
<td>G 3  Gaseous release due to line break caused by a seismic event</td>
<td>NP</td>
<td>F2/S2</td>
<td>TC R-01 OU R-01a</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>PB R-01a</td>
</tr>
<tr>
<td>G 4  Gaseous release due to facility fire</td>
<td>O</td>
<td>F2/S2</td>
<td>WB L-01 PB L-09</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>WB L-02 PB L-10</td>
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<td>WB L-03 PB L-13</td>
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<td>WB L-04 PB L-14</td>
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<td>WB L-06 PB-16</td>
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<td>WB L-07 PB-P-02</td>
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<td></td>
<td>WB L-08 OU P-02</td>
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<tr>
<td><strong>Liquid releases (Section B3 4 2 2)</strong></td>
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<td></td>
</tr>
<tr>
<td>L 1  Spray release due to piping failures</td>
<td>O</td>
<td>F2/S2</td>
<td>PW N-01 PW H-06d</td>
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<tr>
<td></td>
<td>Ei</td>
<td></td>
<td>PW L-02 PW L-14</td>
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<td></td>
<td></td>
<td></td>
<td>PW L-08 PW L-16</td>
</tr>
<tr>
<td>L 2  Spray release due to fire</td>
<td>O</td>
<td>F2/S2</td>
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<td></td>
<td></td>
</tr>
<tr>
<td>L 3  Spray release due to a seismic event</td>
<td>NP</td>
<td>F2/S2</td>
<td>OU R-01b</td>
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<tr>
<td><strong>MCO external hydrogen explosions (Section B3 4 2 3)</strong></td>
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</tr>
<tr>
<td>E 1  Hydrogen explosion outside an MCO due to hydrogen generation within the cask</td>
<td>O</td>
<td>F3/S2</td>
<td>WB B 13g PB L 11a</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>WB B-02a PB L 11f</td>
</tr>
<tr>
<td>E 2  Hydrogen explosion outside an MCO due to instrumentation failure</td>
<td>O</td>
<td>F3/S2</td>
<td>WB B-03b PB B 13d</td>
</tr>
<tr>
<td>E 3  Hydrogen explosion outside an MCO due to excessive water in MCO</td>
<td>O</td>
<td>F3/S2</td>
<td></td>
</tr>
<tr>
<td>E 4  Hydrogen explosion outside an MCO due to process upset of key parameters</td>
<td>O NP Ei</td>
<td>F3/S2</td>
<td>WB B-03a PB H 11d</td>
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<td></td>
<td></td>
<td></td>
<td>WB B 13a PB H 11e</td>
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<td></td>
<td>WB B 13b PB H 11d</td>
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<td>WB B 13c PB H 11e</td>
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<td></td>
<td>WB H-06f</td>
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<td></td>
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<td>WB H-06i</td>
</tr>
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<td>Candidate accident</td>
<td>Initiator type*</td>
<td>Frequency/ consequence categories*</td>
<td>Hazard analysis checklist designator*</td>
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<td>E 5 Hydrogen explosion outside an MCO due to loss of support utilities</td>
<td>O</td>
<td>F3/S2</td>
<td>PB F-02a OU P-04</td>
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<td></td>
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<td></td>
<td>PB F-05 OU R-02</td>
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<td>SB F-01b OU R-03</td>
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<td></td>
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<td></td>
<td>SB F-02b OU R-04</td>
</tr>
<tr>
<td>E 6 Hydrogen explosion outside an MCO due to facility fire</td>
<td>O</td>
<td>F2/S2</td>
<td>PB L-01 PB L-10</td>
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<td>PB L-02 PB L-13</td>
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<td>PB L-07 TC J 12</td>
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<td>PB L-08 OU P-02a</td>
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<td>PB L-09</td>
</tr>
<tr>
<td>E 7 Hydrogen explosion outside an MCO due to contamination of helium supply</td>
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<td>F2/S2</td>
<td>PB H-06k</td>
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<td>E 8 Hydrogen explosion outside an MCO due to line break caused by a seismic event</td>
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<td>PB R-01a</td>
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<td>MCO internal hydrogen explosions (Section B3.4.2.4)</td>
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<td>I 1 Hydrogen explosion within an MCO due to process upset of key parameters (significant air ingress into the MCO)</td>
<td>O</td>
<td>F3/S2</td>
<td>PB B-03a PB H 11d</td>
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<td></td>
<td></td>
<td>PB B 13a PB H 11e</td>
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<td></td>
<td>PB B 13b PB L 11d</td>
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<td>I 2 Hydrogen explosion within an MCO due to instrumentation failure (significant air ingress into the MCO)</td>
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<td>F3/S2</td>
<td>PB B-02a</td>
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<td>F2/S2</td>
<td>PB L-01 PB L-08</td>
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<td>PB L-02 PB L-09</td>
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<td>PB L-03 PB L 10</td>
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<td>PB L-04 PB L-13</td>
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<td>PB L-05 PB L-14</td>
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<td>PB L-06 PB L-15</td>
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<td>PB L-07 PB L-16</td>
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<td>I 4 Hydrogen explosion within an MCO due to hydride reaction</td>
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<td>F0/S2</td>
<td>PB J 12</td>
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<td>I 5 Hydrogen explosion within an MCO due to loss of support utilities (significant air ingress into the MCO)</td>
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<td>F3/S2</td>
<td>PB F-02a OU P-04</td>
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<td></td>
<td></td>
<td>PB F-05 OU R-02</td>
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<td></td>
<td></td>
<td></td>
<td>SB F-01b OU R-03</td>
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<td></td>
<td></td>
<td></td>
<td>SB F-02b OU R-04</td>
</tr>
<tr>
<td>I 6 Hydrogen explosion within an MCO due to line break caused by a seismic event</td>
<td>NP</td>
<td>F2/S2</td>
<td>PB R-01a OU R-01a</td>
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<tr>
<td>Candidate accident</td>
<td>Initiator type$^a$</td>
<td>Frequency/consequence categories$^b$</td>
<td>Hazard analysis checklist designator$^c$</td>
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<tr>
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<td>----------------------------------------</td>
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<tr>
<td>T 1 Thermal runaway reaction due to internal process upset of key parameters$^d$</td>
<td>O NP EI</td>
<td>F3/S3</td>
<td>PB B 13c$^d$ OU R-02</td>
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<td></td>
<td>PB B 13d OU R-03</td>
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<td>PB H-08 OU R-04</td>
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<td></td>
<td>OU P-04</td>
</tr>
<tr>
<td>T 2 Thermal runaway reaction in MCO due to instrumentation failure</td>
<td>O F3/S3</td>
<td></td>
<td>PB B-02b PN B-03a</td>
</tr>
<tr>
<td>T 3 Thermal runaway reaction in MCO due to loss of MCO control caused by facility fire</td>
<td>O F3/S3</td>
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<tr>
<td>T 4 Thermal runaway reaction in MCO due to loss of support utilities</td>
<td>O NP EI</td>
<td>F3/S3</td>
<td>PB F-02a SB F-01b</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>PB F-05 SB F-02b</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>PB H 11f</td>
</tr>
<tr>
<td>T 5 Thermal runaway reaction in MCO due to contamination of helium supply</td>
<td>O F2/S2</td>
<td></td>
<td>PB H-06k</td>
</tr>
<tr>
<td>T 6 Thermal runaway reaction in MCO due to line break caused by a seismic event</td>
<td>NP F2/S3</td>
<td></td>
<td>PB H-06k</td>
</tr>
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<tr>
<td>MCO overpressurization (Section B3 4 2 6)</td>
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<td></td>
</tr>
<tr>
<td>P 1 Overpressurization due to internal process upset of key parameters$^d$</td>
<td>O F3/S3</td>
<td></td>
<td>PB B-03a OU R-02</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>PB B 13c$^d$ OU R-02</td>
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<td>PB B 13d OU R-04</td>
</tr>
<tr>
<td>P 2 Overpressurization due to loss of support utilities</td>
<td>O NP EI</td>
<td>F2/S3</td>
<td>PB F-02a SB F-01b</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>SB F-01b OU P-04</td>
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<td></td>
<td></td>
<td></td>
<td>OU R-03</td>
</tr>
<tr>
<td>P 3 Overpressurization due to excessive helium supply pressure</td>
<td>O F2/S2</td>
<td></td>
<td>PB H 11a</td>
</tr>
<tr>
<td>P 4 Overpressurization due to a line break caused by a seismic event</td>
<td>NP F2/S3</td>
<td></td>
<td>PB R-01a OU R-01a</td>
</tr>
</tbody>
</table>
### Table B3-4  Binned Listing of Candidate Accidents Sorted by Risk Ranking  (4 sheets)

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Initiator type</th>
<th>Frequency/ consequence categories&lt;sup&gt;a&lt;/sup&gt;</th>
<th>Hazard analysis checklist designtor&lt;sup&gt;b&lt;/sup&gt;</th>
</tr>
</thead>
<tbody>
<tr>
<td>P 5 Overpressurization caused by facility fire</td>
<td>O</td>
<td>F3/S3</td>
<td>PB L-01 PB L-09</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>PB L-02 PB L 10</td>
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<td></td>
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<td>PB L-03 PB L 13</td>
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<td>PB L-04 PB L 14</td>
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<td>PB L-05 PB L 15</td>
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<td>PB L-06 PB L 16</td>
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<td>PB L-07 PB P-02</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>PB L-08 OU P-02a</td>
</tr>
</tbody>
</table>

<sup>a</sup>O = operational  NP = natural phenomena  EI= externally intuated

<sup>b</sup>S2 = Sufficient material and release energy to affect an onsite receptor (collocated worker) 100 m from the source of the release
S3 = Sufficient material and release energy to affect a receptor at the nearest point of uncontrolled public access (site boundary)
F0 = Beyond extremely unlikely (not credible)
F2 = Foreseeable but unlikely
F3 = Likely to occur during the lifetime of the facility

Hazard analysis checklist designators are from, HNF SD SNF HIE-004 1999 Cold Vacuum Drying Facility Hazard Analysis Report Rev 4 Fluor
Danzel Hanford, Incorporated, Richland, Washington  Figure B3 2 shows the hazardous material/energy source checklist that was used to group potentially hazardous materials and energy sources for each of the six major building areas
AA Administrative area
TC Transfer corridor and mechanical corridor
PB Process bays 2 through 5
SB Process bay 1 an unused bay
PW Process water tank room
OU Outside
An alpha numeric designator from the standardized checklist is combined with the building identifier to provide a unique identifier for each hazard
(e g AA A-01 identifies an electrical source in the administrative area specifically battery banks)  If a hazard is evaluated in the hazard analysis tables (HNF SD SNF HIE-004) using multiple rows a lowercase letter following the designator is used to differentiate between rows (e g AA A-01a AA A-01b)

Chosen as a representative and bounding accident for further accident analysis development

Normally F0 events are excluded from the accident bins but the hydride reaction as an independent initiator is identified explicitly to demonstrate that it has been considered. Hydride reactions are included in the calculations for other DBAs

CVDF = Cold Vacuum Drying Facility
HVAC = heating, ventilation, and air conditioning
MCO = multi-canister overpack
descriptions, and operation and maintenance procedures. Abnormal events encompass malfunctions of systems, operating upset conditions, equipment failures, safety system trips, natural phenomena or operator error that may impact operational or programmatic schedules. While many postulated CVDF abnormal events have similar initiators as the DBAs and other accidents analyzed in Section B3.4, there are no radiological releases outside the facility. Facility worker exposure to radiological hazards associated with abnormal events are controlled by Hanford Site, SNF Project, and CVDF radiological protection and ALARA programs and practices.

The CVDF abnormal event analysis was developed from a review of HNF-SD-SNF-HIE-004 which identified hazardous conditions and their potential accidents or events meeting the abnormal event profile. Hazardous conditions having common failures or impacts on facility operations were then grouped into categories. The following five abnormal event categories, each having distinct initiators and operational consequences, were identified for the CVDF process:

1. MCO received in a suspect condition
2. Single or multiple, empty process bays out-of-service
3. Single, processing process bay loss of processing ability
4. Shipping delays from the CVDF
5. Safe and stable state following safety system actuation

The following sections provide a description of these five CVDF abnormal event categories. For each of the abnormal event categories, the descriptive information includes the event, the cause of the event, the consequence of the event, the means of detection for the event, and corrective actions. After the safe and stable condition is achieved for each type of event, any necessary recovery actions or plans must be developed based on the specific abnormal condition. All applicable requirements must be followed in the recovery actions or plans (e.g., reviewed safety question evaluations).

B3.3.1 Multi-Canister Overpack Received in a Suspect Condition

- **Event** An MCO shipped from the K Basins is received in a suspect or indeterminate condition at the CVDF. Events in this category have a potential impact on CVDF operations by causing delays due to the uncertain chemical or thermal status of the SNF in the MCO. MCO transportation requirements are established in the safety analysis report for packaging (HNF-SD-TP-SARP-017), which requires transport from the K Basin to the CVDF within a shipping window. Events occurring during transport are analyzed in the safety analysis report for packaging. Cask-MCOs experiencing unusual transportation conditions and received and accepted by the CVDF would require evaluation prior to processing.

- **Postulated Cause of Event** Events in this category involve (1) receiving an MCO with a transport time greater than the shipping window, (2) receiving an MCO that
experienced unusual transport conditions, (3) receiving an MCO with higher than expected pressure, or (4) receiving an MCO with an incomplete or irregular quality assurance package

- **Detection of Event**  Events in this category are detected by operator observations or direct readings from pressure test instrumentation. A higher than expected pressure reading requires a special venting procedure.

- **Analysis of Effects and Consequences**  Events in this category could impact the facility or project schedules and may result in an increase of facility occupational exposure totals. Receiving an out-of-specification MCO would likely require an investigation and could cause facility delays, if special venting or processing requirements were identified and imposed. While the facility exposure totals and ALARA goals may be impacted, individual workers would not exceed radiological protection limits while responding to events in this category.

- **Corrective Actions**  Immediate operator actions to stabilize events in this category are contained in standard operating procedures, alarm response procedures, and/or emergency response guides. Once the event has been stabilized, facility management will develop a recovery plan based on the existing facility conditions, the cause of the abnormal event, and safety requirements.

### B3 3 2  Single or Multiple, Empty Process Bays Out of Service

- **Event**  One or more process bays at the CVDF are out of service. Events in this category impact CVDF operations because conditions cause one or more empty operating bays to be out of service and unable to receive an MCO and cask from the K Basins. Until the equipment or structures are repaired or returned to operating status, the bay or bays are unable to receive an MCO.

- **Postulated Cause of Event**  Conditions causing a bay or bays to be declared out of service could result from electrical or mechanical failure, structural damage, fire, or operator errors.

- **Detection of Event**  Events in this category are detected by instrument readings, direct observation of equipment failure by operator or maintenance personnel, and visual or audible alarms.

- **Analysis of Effects and Consequences**  Events in this category primarily result in schedule delays for the CVDF and the potential for programmatic delays to the SNF Project. Equipment failures can lead to facility delays for repairs and tests and additionally, possible delays for investigations to resolve the reasons for the equipment failures and to determine future requirements.
HNF-3553 REV 0  
Annex B — Cold Vacuum Drying Facility

- **Corrective Actions**  Immediate operator actions to bring events in this category to operational safe and stable states are contained in alarm response procedures and/or emergency response guides. In most cases, equipment operations during recovery are expected to be the same as those contained in the standard operating and/or maintenance procedures.

**B3 3 3 3 Single, Processing Process Bay Loss of Processing Ability**

- **Event**  A process bay loses the ability to continue processing an MCO. Events in this category impact CVDF operations by delaying processing in the affected bay. Depending on the cause and duration of the event, an impact to programmatic schedules may occur. Until the equipment or structure is repaired or returned to operating status, the MCO is maintained in a safe and stable condition identified in alarm response procedures.

- **Postulated Cause of Event**  Conditions affecting the ability to process an MCO in a single bay could result from electrical or mechanical failure, structural damage, fire, or operator errors.

- **Detection of Event**  Events in this category are detected by instrumentation readings, direct observation of equipment failure by operator or maintenance personnel, visual or audible alarms, or notification from Hanford Site or offsite entities.

- **Analysis of Effects and Consequences**  Events in this category primarily result in schedule delays for the CVDF, the potential for programmatic delays to the SNF Project, and the potential need to perform unreviewed safety question evaluations to assess delay impacts to safety analysis assumptions before resumption of processing. Equipment failures can lead to facility delays for repairs and tests, and additionally, possible delays for investigations to resolve the reasons for the equipment failures and to determine future requirements. In the event of a fire, equipment and personnel could be impacted by smoke, heat, and chemical vapors.

- **Corrective Actions**  Immediate operator actions to bring events in this category to operational safe and stable states are contained in alarm response procedures and/or emergency response guides. Once the event has been stabilized, facility management will develop a recovery plan based on the existing facility conditions, the cause of the abnormal event, and safety requirements. In most cases, equipment operations during recovery are expected to be the same as those contained in the standard operating and/or maintenance procedures.
B3 3 3 4 Shipping Delays from the Cold Vacuum Drying Facility

- **Event**  An MCO cannot be shipped from the CVDF and remains in the process bay. Events in this category occur following the drying process and impact CVDF operations by reducing the availability of processing bays for receipt of cask-MCOs from K Basins. Since the drying process has been completed, the affected MCO is in a safe and stable condition.

- **Postulated Cause of Event** Conditions affecting the ability to ship the cask-MCO could result from electrical or mechanical failure, structural damage, fire problems at the CSB, weather delays, or operator errors.

- **Detection of Event** Events in this category are detected by instrumentation readings, direct observation by operating personnel, visual or audible alarms, or by notification from Hanford Site or offsite entities.

- **Analysis of Effects and Consequences** Events in this category primarily result in schedule delays for the CVDF and the potential for programmatic delays to the SNF Project, and exceeding the shipping window to the CSB. Equipment failures can lead to facility delays for repairs and tests, and additionally, possible delays for investigations to resolve the reasons for the equipment failures and to determine future requirements. In the event of a fire, equipment and personnel could be impacted by smoke, heat, and chemical vapors.

- **Corrective Actions** Immediate operator actions to bring events in this category to operationally safe and stable states are contained in standard operating procedures, alarm response procedures, and/or emergency response guides. Once the event has been stabilized, facility management will develop a recovery plan based on the existing conditions, the cause of the abnormal event, and safety requirements.

B3 3 3 5 Safe and Stable State following Safety System Actuation

- **Events** CVDF safety systems are actuated, which halts normal MCO processing. Events in this category are the safe and stable states established as a result of automatic or manual safety-class instrumentation and control (SCIC) system actuation. Facility and programmatic delays occur as a result of these events ranging from the time necessary to return the system to normal processing to the time necessary to investigate the event cause and institute corrective measures.

- **Postulated Cause of Event** Events in this category occur as a result of (1) SCIC system automatic protective action, if monitored process or facility parameters reach predetermined limits, or (2) operating personnel manually tripping the safety-class helium (SCHe) system in response to a process upset, alarm response procedure, or
emergency response procedure  Potential initial causes for the automatic or manual protective action are the analyzed accidents described in Section B3.4.2

- **Detection of Event**  SCIC system and SCHe system actuation are alarmed in the facility control room

- **Analysis of Effects and Consequences**  SCIC system and SCHe system actuation occur to prevent and mitigate events before unacceptable conditions are reached  These systems place a single bay, MCO, or all MCOs in the facility in safe and stable states depending on the event  Events in this category primarily result in schedule delays for the CVDF, with the potential for programmatic delays to the SNF Project  SCIC automatic actuation may require an investigation, reporting and possible corrective measures  Manual SCHe actuation would require similar activities

- **Corrective Actions**  Immediate operator actions to bring events in this category to operational safe and stable states are contained in alarm response procedures and/or emergency response guides  Operating personnel are routinely trained and qualified in the use of these procedures and guides  Once the event has been stabilized facility management will develop a recovery plan based on the existing facility conditions, the cause of the SCIC actuation, and safety requirements  In most cases equipment operations during recovery are expected to be the same as those contained in the standard operating and/or maintenance procedures

**B3.4 ACCIDENT ANALYSIS**

This section presents the methodology used to analyze the DBAs identified in Table B3-4, and the quantified results of those analyses  It also presents the safety-class and safety-significant SSCs and TSRs related to these accident events that are necessary for the protection of the offsite public and onsite workers  For each DBA, the following standard topics are discussed (DOE-STD-3009-94)

- **Scenario development**

  Accident analysis for each DBA starts with a formal description of the accident scenario  Each description is supported by a basic event tree (SNF-2770)  The major assumptions in each scenario are identified

- **Source term analysis**

  The source term for each accident is obtained through phenomenological and system response calculations
Consequence analysis

Atmospheric dispersion or other relevant pathways of concern are used in calculating the consequences of each DBA. Because the potential consequences of the CVDF DBAs are significant, computer modeling using site-specific meteorological data was used to model the dispersion.

Comparison with guidelines

The consequences determined for each DBA are compared with release limits and risk evaluation guidelines. From this, it is determined whether safety-class SSC designation is needed. The need for accident-specific TSRs to meet release limits and risk evaluation guidelines is determined.

Summary of safety SSCs and TSRs

The safety SSCs and TSRs are summarized. Detailed descriptions of safety-class SSCs are presented in Chapter B4.0, and detailed descriptions of the TSRs are presented in Chapter B5.0.

In addition to the DBAs analyzed in this section, common cause (facility-wide) events are discussed in Section B3.4.2.7, and beyond design basis accidents (BDBAs) are discussed in Section B3.4.3.

The DBAs are summarized below and are developed in greater detail in Section B3.4.2.

Gaseous release (Section B3.4.2.1)

The bounding scenario for this accident category describes a pressurized release of helium gas and entrained contaminated particulate through a process line leak. The unmitigated consequences of this event do not exceed the offsite release limits but do exceed the onsite risk evaluation guidelines. No safety-class features are required to mitigate this event. Safety-significant features selected for this event include portions of the process general supply/exhaust heating, ventilation, and air conditioning (HVAC) system, the process bay local exhaust HVAC and process vent system (including backup electrical power supplied by the standby power system) and the process bays and process water tank room differential pressure alarms. Mitigated consequences of this event are well below both offsite release limits and onsite risk evaluation guidelines. Refer to Section B3.4.2.1, Table B3-9, for a detailed list of safety features selected to prevent, mitigate, or provide defense in depth for the accident scenarios binned in the gaseous release accident category.
Liquid release (Section B3 4 2 2)

The bounding scenario for this accident category describes a pressurized leak of water and entrained contaminated particulate from the PWC piping. The unmitigated consequences of this event do not exceed the offsite release limits but do exceed the onsite risk evaluation guidelines. No safety-class features are required to mitigate this event. Safety-significant features selected for this event include portions of the process general supply/exhaust HVAC system (duct work and HEPA filters for process water tank room) and the process water tank room differential pressure alarm. Mitigated consequences of this event are well below both offsite release limits and onsite risk evaluation guidelines. Refer to Section B3 4 2 2, Table B3-12, for a detailed list of safety features selected to prevent, mitigate, or provide defense in depth for the accident scenarios binned in the liquid release accident category.

MCO external hydrogen explosion (Section B3 4 2 3)

The bounding scenario for this accident category describes accumulation of hydrogen outside an MCO when it is vented from the MCO into the local exhaust process ventilation system and mixed with air, followed by ignition and explosion of the hydrogen gas. The unmitigated consequences of this event do not exceed the offsite release limits but do exceed the onsite risk evaluation guidelines. No safety-class features are required to prevent or mitigate this event. Safety-significant features selected to prevent this event include portions of the process bay local exhaust HVAC and process vent system (ensuring minimum flow rate), a flow-restricting orifice to limit the cask vent flow rate, and the standby power system. Potential releases are mitigated by HEPA filter loading limits. Because the selected safety features prevent and mitigate this event both offsite release limits and onsite risk evaluation guidelines are satisfied. Refer to Section B3 4 2 3, Table B3-14, for a detailed list of safety features selected to prevent, mitigate, or provide defense in depth for the accident scenarios binned in the external hydrogen explosion accident category.

MCO internal hydrogen explosion (Section B3 4 2 4)

The bounding scenario for this accident category describes the ignition and explosion of a hydrogen-air mixture inside an MCO. The unmitigated consequences of this event do not exceed the offsite release limits but do exceed the onsite risk evaluation guidelines. No safety-class features are required to prevent or mitigate this event. However, some safety-class features preventing the MCO thermal runaway reaction and overpressurization events (i.e., multiple safety features to detect process upsets, the SCHe system, portions of the tempered water [annulus] system, and the water isolation components) also prevent this accident, but in a safety-significant role. The safety-significant portions of the process bay local exhaust HVAC (including backup...
power to the process bay local exhaust HVAC from the standby power system) and the process general exhaust HVAC systems also provide a mitigation function, which represents an added safety barrier if any of the prevention features fail. Because the designated safety features prevent and mitigate this event, both offsite release limits and onsite risk evaluation guidelines are satisfied. Refer to Section B3 4 2 4, Table B3-16, for a detailed list of safety features selected to prevent, mitigate, or provide defense in depth for the accident scenarios binned in the internal hydrogen explosion accident category.

- **MCO thermal runaway reaction (Section B3 4 2 5)**

The bounding scenario for this accident category describes an accident that is initiated by a reduction of heat removal from the MCO. This condition results in escalation of the chemical reaction within the MCO and high temperatures in the MCO. The high temperatures in the unmitigated scenario ultimately lead to a release of gas and radioactive particulate for an extended period of time. The release is caused by rapid uranium-water reactions inside the MCO. The unmitigated consequences of this event exceed the offsite release limits and exceed the onsite risk evaluation guidelines. Safety-class features selected to prevent this event include portions of the tempered water (annulus) system to provide adequate heat transfer from the MCO, a TSR, and associated equipment to refill the annulus if annulus water is lost. The SCHe system, and safety-class instrumentation to detect process upsets and actuate the SCHe system, act in a defense-in-depth role by providing more time before a thermal excursion occurs (purge and vent function). The estimated frequency for failure of the preventive features is in the beyond extremely unlikely category (less than $10^{-6}$ per year). Refer to Section B3 4 2 5, Table B3-18, for a detailed list of safety features selected to prevent, mitigate, or provide defense in depth for the accident scenarios binned in the MCO thermal runaway accident category.

- **MCO overpressurization (Section B3 4 2 6)**

The bounding scenario for this accident category describes an overpressurization of an isolated MCO with no pressure relief. The pressure in an isolated MCO increases with the formation of hydrogen gas as a product of the uranium-water reaction. The MCO internal pressure would continue to increase until the MCO pressure boundary is breached or until the fuel or water are completely consumed. The overpressurization leads to a pressurized release of gas and radioactive particulate followed by an extended period of a slow continuous release driven by the continued oxidation of the uranium inside the MCO. The unmitigated consequences of this event exceed both the offsite release limits and the onsite evaluation risk guidelines. Safety-class features selected to prevent this event include multiple safety features to detect process upsets, the SCHe system, the 30 lb/in² gauge rupture disk and vent line, the 150 lb/in² gauge rupture disk, and portions of the tempered water (annulus).
system These safety-class features reduce the frequency and mitigate the occurrence of this event to well within the offsite release limits. Additional safety significant features for confinement and filtration are identified to mitigate the onsite consequences to well below the onsite risk evaluation guidelines. Refer to Section B3 4 2 6, Table B3-22, for a detailed list of safety features selected to prevent, mitigate, or provide defense in depth for the accident scenarios binned in the MCO overpressurization accident category.

In a small number of cases, operator actions are credited as TSR items for response to upset conditions. In all of these cases, the credited actions are simple and can easily be accomplished in the time available. In addition, credit for such actions is minimized and only taken in cases where the risk is already low (frequencies or consequences are small).

Letter 97-SFD-172, Risk Evaluation Guidelines (REGs) to Ensure Inherently Safer Designs (Sellers 1997), provides guidance for determining the need for safety-class and safety-significant SSCs. Table B3-5 is extracted from the SNF Project FSAR and summarizes that guidance for radiological consequences. If a credible event can cause an offsite radiation exposure exceeding 0.5 rem, then safety-class SSCs must be provided in the design to prevent or mitigate the consequences of the event. Similarly, if a credible event can cause an onsite radiation exposure, then safety-significant SSCs must be provided to prevent or mitigate the consequence. If failure of a credited safety-class system is credible, the offsite consequences must be limited to the value in Table B3-5 for the appropriate likelihood category.

### Table B3-5 Radiological Evaluation Guidelines and Limits

<table>
<thead>
<tr>
<th>Event category</th>
<th>Frequency range (per year)</th>
<th>Onsite risk evaluation guidelines* (rem)</th>
<th>Offsite accident release limits* (rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Anticipated</td>
<td>1.0E-01 to 1.0E-02</td>
<td>1</td>
<td>0.5</td>
</tr>
<tr>
<td>Unlikely</td>
<td>1.0E-02 to 1.0E-04</td>
<td>10</td>
<td>50</td>
</tr>
<tr>
<td>Extremely unlikely</td>
<td>1.0E-04 to 1.0E-06</td>
<td>25</td>
<td>50</td>
</tr>
</tbody>
</table>

Note: All doses are committed effective dose equivalents.

*This terminology is consistent with Tables 1 and 2 of Sellers E D 1997 Risk Evaluation Guidelines (REGs) to Ensure Inherently Safer Designs (Letter 97 SFD 172 to H J Hatch Fluor Daniel Hanford Incorporated August 26) U S Department of Energy Richland Operations Office Richland Washington.

The consequences associated with each of these six bounding DBAs are summarized in Table B3-6.
**Table B3-6  Summary of Consequences for Bounding Design Basis Accidents**

<table>
<thead>
<tr>
<th>Event</th>
<th>Section</th>
<th>Offsite consequences*</th>
<th>Onsite consequences</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Release limit</td>
<td>Unmitigated</td>
</tr>
<tr>
<td></td>
<td></td>
<td>(rem)</td>
<td>(rem)</td>
</tr>
<tr>
<td>Gaseous release</td>
<td>B3 4 2 1</td>
<td>0.05</td>
<td>1.7E-03</td>
</tr>
<tr>
<td>Liquid release</td>
<td>B3 4 2 2</td>
<td>0.05</td>
<td>1.2E-02</td>
</tr>
<tr>
<td>MCO external hydrogen explosion</td>
<td>B3 4 2 3</td>
<td>0.05</td>
<td>8.6E-03</td>
</tr>
<tr>
<td>MCO internal hydrogen explosion</td>
<td>B3 4 2 4</td>
<td>0.05</td>
<td>0.29</td>
</tr>
<tr>
<td>MCO thermal runaway reaction</td>
<td>B3 4 2 5</td>
<td>0.05</td>
<td>0.59</td>
</tr>
<tr>
<td>MCO overpressurization</td>
<td>B3 4 2 6</td>
<td>0.05</td>
<td>0.70</td>
</tr>
</tbody>
</table>

Note: Offsite release limits and onsite risk evaluation guidelines cited are for the "anticipated event frequency category as provided in Sellers E D 1997 Risk Evaluation Guidelines (REGS) to Ensure Inherently Safer Designs (Letter 97 SFD 172 to H J Hatch Fluor Daniel Hanford Incorporated August 26) U S Department of Energy Richland Operations Office Richland Washington

*NRC equivalency for consideration of organ doses is addressed in HNF SD SFN TI-059 1999 A Discussion on the Methodology for Calculating Radiological and Toxicological Consequences for the Spent Nuclear Fuel Project at the Hanford Site Rev 2 Fluor Daniel Hanford Incorporated Richland Washington

EG = evaluation guideline
MCO = multi-canister overpack

**B3 4 1 Methodology**

This section identifies CVDF-specific methods and assumptions used to quantify the consequences of the DBAs. Methods and assumptions that are common or generic to all the SNF Project facilities at the K Basins, CVDF, or CSB are described in Chapter 30 of the SNF Project FSAR.

**B3 4 1 1 Source Term** The bounding source term used for the accident analysis is based on data for the SNF in the K East and K West Basins that is provided in HNF-SD-SNF-TI-009, 105-K Basin Material Design Basis Feed Description for Spent Nuclear Fuel Project Facilities. This document defines an inventory for safety analysis based on the inventories of Mark IV and Mark IA fuel in the K Basins. High-burnup Mark IV fuel, the fuel type that results in the highest estimated dose per gram, was selected as the bounding inventory from the radiological dose perspective. Nuclear accountability records provide the basis for the quantity, exposure variation, and decay time variation of the stored fuel. The radionuclide inventory has been estimated from these data (HNF-SD-SNF-TI-009).
The MCO contains finely divided particulate material associated with the oxidation of the SNF. This material includes an oxide layer on the fuel and particulate remaining on fuel surfaces and in crevices after fuel washing and racking into the MCO as well as oxidation products that are created during queuing at the K Basins and processing at the CVDF. The particulate inventory of the MCO dominates the airborne release.

For both the onsite and offsite receptors at all accident frequency categories, the radiological guidelines are more limiting than the toxicological guidelines for the release of SNF particulate. Therefore, the toxicological consequences of the postulated airborne releases do not require mitigating features beyond those required by the radiological consequences (see Section 3.4.1 of the SNF Project FSAR).

No routine chemical processes will be conducted within the CVDF. Purging and backfilling the MCOs will involve the use of an inert gas. Some chemicals, such as those used for equipment decontamination, may be used occasionally (HNF-SD-SNF-CM-001). However, there are no chemical inventories of concern for safety analysis considerations.

**B3 4.1.2 Consequence Analysis** Radiological inhalation dose consequences are calculated for four receptor locations:

- **Hanford Site boundary (10,090 m [33,100 ft])** — release limits defined, used for calculation of offsite doses and selection of safety-class features.

- **Collocated worker at 100 m (328 ft)** — risk evaluation guidelines defined, used for calculation of onsite doses and selection of safety-significant features.

- **Near bank of the Columbia River (onsite, at a distance of approximately 650 m [2,130 ft])** — no evaluation guideline defined, doses calculated for the purpose of identifying any additional measures considered necessary to reduce the dose to individuals at this receptor location.

- **100 Area Fire Station (onsite, at a distance of approximately 3,750 m [12,300 ft])** — no evaluation guideline defined, doses calculated for the purpose of identifying any additional measures considered necessary to reduce the dose to individuals at this receptor location.

Radiological inhalation dose consequences for each accident analyzed are based on the following factors:

- **Mass of material at risk (MAR) available for release, g**

- **Airborne release fraction (ARF) (or the product of the airborne release rate [ARR] and release time [T]) and respirable fraction (RF)**
• Leak path factor (LPF)

• Mass of respirable material released (M)

• Atmospheric transport factor (\(x/Q'\)), s/m³

• Breathing rate (BR), m³/s

• Dose per unit intake (UD) of SNF, rem/g (Sv/g)

• Duration of exposure

• Committed effective dose equivalent (D), rem (Sv)

The quantity of respirable material released (M) is determined by the specific accident scenario. The quantity M is a function of the MAR, total ARF (or ARR times T), the RF, and the leak path factor of any passive structural enclosure that may cause deposition of an airborne release before the release enters the atmosphere. The leak path factor is based on a time-integrated calculation of aerosol deposition within and release from an enclosure of given dimensions with specified leakage area, pressure, and temperature differentials. The specific value of each parameter is determined in the individual DBA analysis and based on the physical phenomena of the accident, thus the values are specific to the CVDF.

The atmospheric transport factor (\(x/Q'\)) is based on specific release conditions (e.g., ground level or elevated, long or short duration) and the receptor's distance from the release. While the methodology is common to the SNF Project, the atmospheric transport factor is the time-integrated normalized air concentration at the receptor's location, which is a measured distance from the CVDF. The transport factor includes the dilution of an airborne contaminant caused by atmospheric mixing and turbulence. The air transport values used in this report have been generated using the GXQ computer program (WHC-SD-GN-SWD-30002 WHC-SD-GN-SWD-30003). Table B3-7 contains the air transport values used to determine onsite and offsite consequences.

Air transport factors were calculated using methods found in NRC Regulatory Guide 1 145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.

Exposures to the collocated worker onsite are bounded by exposures of the individual at the 100-m location. The risk evaluation guidelines apply to this individual (Sellers 1997). For assessment purposes, DOE has directed (Sellers 1996) that the Hanford Site boundary be considered the location of the offsite receptor. Consequences at the near bank of the Columbia River are provided for information related to evaluation of possible additional controls applied to protect receptors at this location.
Table B3-7  Atmospheric Transport Factors Used in Accident Analyses for the Cold Vacuum Drying Facility

<table>
<thead>
<tr>
<th>Receptor location description</th>
<th>Air transport factors for various release durations(^a)</th>
<th>Acute</th>
<th>Meander</th>
<th>Logarithmic interpolation(^b)</th>
<th>Chrome</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Less than 1 hour(^c)</td>
<td>1 to 2 hours(^d)</td>
<td>12 hours</td>
<td>24 hours</td>
</tr>
<tr>
<td>Onsite (100 m E)</td>
<td></td>
<td>7.32 E-02</td>
<td>1.24 E-02</td>
<td>6.28 E-03</td>
<td>NA</td>
</tr>
<tr>
<td>Columbia River near bank (650 m W)</td>
<td></td>
<td>2.44 E-03</td>
<td>4.25 E-04</td>
<td>1.99 E-04</td>
<td>NA</td>
</tr>
<tr>
<td>100 Area Fire Station (3 750 m ESE)</td>
<td></td>
<td>1.60 E-04</td>
<td>7.82 E-05</td>
<td>2.73 E-05</td>
<td>NA</td>
</tr>
<tr>
<td>Hanford Site boundary (10 090 m W)</td>
<td></td>
<td>4.48 E-05</td>
<td>3.11 E-05</td>
<td>1.01 E-05</td>
<td>6.50 E-06</td>
</tr>
</tbody>
</table>

*Units for these values are seconds per cubic meter. In all cases the releases are assumed to be point sources at ground level to maximize the dose consequences.

*Air transport factors are computed by logarithmic interpolation between the 1 to 2 hours value and the annual average value.

*No adjustment for plume meander.

*Plume meander adjustment used is found in NRC Regulatory Guide 1 145 1982 *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants* U.S. Nuclear Regulatory Commission Washington D.C.

*The 12 hour durations are used for onsite exposures and the 24 hour durations are used for offsite exposures.

(HNF SD SNF T1 059 1999 *A Discussion on the Methodology for Calculating Radiological and Toxicological Consequences for the Spent Nuclear Fuel Project at the Hanford Site* Rev 2 Fluor Daniel Hanford Incorporated Richland Washington) These are bounding exposure times.

NA = not applicable

If the offsite receptor were assumed to be coincident with the receptor at the Columbia River location, it is expected that no additional controls would be imposed and impacts to existing controls would be minimal. For the gaseous release, liquid release, and external explosion DBAs, dose consequences would still be below release limits, so no impacts to controls would be realized. The internal explosion DBA consequences would increase from safety significant to safety class, but many of the credited systems for internal explosion are already safety class because of the thermal runaway and overpressurization DBAs. The thermal runaway and overpressurization DBAs are already safety class and would not be affected.

None of the accidents analyzed in this document adjusts the air transport factors for building wake effects or for the elevation of the release above ground level (the stack height is not sufficient to be used for atmospheric dispersion). It is conservative to ignore building wake and stack effects because both of these effects serve to disperse the release such that the collocated worker and the offsite receptor receive a lower estimated dose, if the effect is included. Section 1 4 1 2 8 of the SNF Project FSAR provides additional information on the calculation of.
the air transport factors. The basis for defining the location of the onsite and offsite receptors is provided in Section 13.13.

The breathing rate (BR) depends on individual exertion factors and exposure duration. This methodology is common to the SNF Project facilities and is described in Chapter 3.0 of the SNF Project FSAR.

The dose per unit intake (UD) is the 50-year dose commitment for all relevant exposure pathways per gram of radioactive material inhaled (HNF-SD-SNF-TI-059). The major radiation exposure pathway for the identified accidents is inhalation of radioactive material. This methodology is common to the SNF Project facilities and is described in Chapter 3.0 of the SNF Project FSAR.

B3.4.1.3 Frequency Estimates. The determination of frequency estimates is based on methodology common to the SNF Project facilities and is described in Chapter 3.0 of the SNF Project FSAR. The CVDF accident frequencies are documented with the DBA development descriptions in Section B3.4.2.

Calculated accident frequencies for each of the six DBAs are presented in Appendix A of SNF-2770. The frequencies for both the unmitigated and the mitigated consequences for a given DBA are presented. An unmitigated event sequence that has a calculated frequency of $10^{-6}$ per year or less does not need further consideration. Unmitigated event sequences with calculated frequencies greater than $10^{-6}$ per year must be evaluated against the offsite release limits and appropriate onsite risk evaluation guidelines in Table B3-5.

B3.4.1.4 Risk Guidelines. The DOE-recommended radiological risk evaluation guidelines (Sellers 1997) are applied across the SNF Project and are described in Chapter 3.0 of the SNF Project FSAR. Satisfaction of the SNF Project radiological evaluation guidelines will meet the goals of SEN-35-91, Nuclear Safety Policy, as discussed in Chapter 3.0.

B3.4.1.5 Safety Structures, Systems, and Components. "Safety class" and "safety significant," as related to the SSCs are defined consistently for the SNF Project in Chapter 3.0 of the SNF Project FSAR.

For each accident scenario, the violation of criticality controls and the airborne radiological dose calculated using the methods described here and in Chapter 3.0 of the SNF Project FSAR, are compared with the appropriate onsite and offsite evaluation guidelines and release limits from Letter 97-SFD-172 (Sellers 1997). If the radiological dose for the unmitigated case exceeds the guideline, mitigative or preventive safety features with appropriate safety-class and/or safety-significant functional classifications, are identified. The dose consequences are recalculated taking appropriate credit for the mitigating safety features to verify that the mitigated doses satisfy the guidelines.
The controls tables at the end of each DBA (e.g., Table B3-9) provide the safety SSCs and administrative features for the evaluated accidents. In addition, other defense-in-depth features are listed. The tables may be used to evaluate with cost-benefit analyses any future improvements in any of these features or safety systems in order to improve operations, maintenance, or design by using the tables' general arrangement of most important systems and features to least important.

The controls tables do not list operability TSRs related to supporting the credited SSCs because their designation is already dictated by specification of the SSCs. For example, by crediting an SSC for a specific function, it is given that the SSC must be available and operable. Such TSRs were not listed to avoid increasing the length of the tables and removing focus from unique TSRs and other controls.

In the selection of controls, general-services SSCs are considered as candidates for providing a safety function if the performance of the general-service SSC is monitored by a safety-related SSC. In these cases, if the general-service SSC fails to provide the necessary function, the safety-related monitoring allows compensatory measures to be taken. For example, the general-service helium system provides a supply of helium, and safety-class flow instrumentation monitors its performance. If the general-service helium system does not function properly, compensatory measures are taken. The CVDF provides a similar approach for confinement function general-service fans with safety-significant HEPA filters provide the confinement function, and the differential pressure from the process bay to the environment has safety-significant monitors to demonstrate the availability of the filtered ventilation. If the differential pressure is lost, confinement cannot be assured and compensatory measures are taken.

B3 4 2 Design Basis Accidents

The results of the DBA analyses that have been performed for the CVDF are summarized in this section along with the key assumptions used in those analyses. The DBAs summaries are based on the guidelines provided in DOE-STD-3009-94, and include the following DBA categories:

- Gaseous release (Section B3 4 2 1)
- Liquid release (Section B3 4 2 2)
- MCO external hydrogen explosion (Section B3 4 2 3)
- MCO internal hydrogen explosion (Section B3 4 2 4)
- MCO thermal runaway reaction (Section B3 4 2 5)
- MCO overpressurization (Section B3 4 2 6)

The DBAs have been analyzed to quantify consequences, and those consequences have been compared with the release limits for offsite consequences and evaluation guidelines for onsite consequences. The process is iterative, starting by taking no credit for mitigative features and comparing the results to the limits or guidelines. Credit is then taken for safety SSCs that
prevent or mitigate the consequences to show that the results are below the release limits and evaluation guidelines. The process continues after the evaluation guidelines are met by identifying other SSCs that, while not designated as safety class or safety significant, provide additional mitigative features as defense in depth. In addition, each individual DBA section references a supporting calculational note (SNF-2770) that provides more detail.

The following assumptions have some relevance to all the DBAs:

- The bounding MCO is loaded at the K Basins with two scrap baskets and three Mark IV fuel baskets. The scrap baskets are at the top and bottom of the MCO. The scrap has an exposed surface area (uranium fuel area without cladding, including indentations, cracks, and crevices exposed to MCO gases or liquids) of 45,000 cm² per scrap basket. Each of the three fuel baskets contains 54 assemblies, and the fuel has an exposed surface area of 7,900 cm² per fuel basket. The total exposed surface area in the bounding MCO is 120,000 cm² (HNF-SD-SNF-TI-015).

- After the fuel is washed, the baskets are queued and loaded into the MCO, transported to the CVDF, and prepared for draining. At this point in the process, the bounding value for the mass of particulate in the bounding MCO is 15 kg in the form of UO₂. Normal values for the mass of particulate in an MCO are expected to be less than 0.1 kg. The complete processing of the MCO at the CVDF produces an additional 10 kg of particulate (SNF-2770, Chapter 10) for a total of 25 kg of UO₂.

- Hydrogen gettering by uranium, and hydrogen production from radiolysis during the processing time interval, is negligible (HNF-SD-SNF-CN-006) and is ignored in this analysis.

**B3 4 2 1 Gaseous Release** This section addresses the consequences of a pressurized release of helium gas and entrained contaminated particulate through a process leak. The quantity of radioactive material released is calculated, and this value is used to calculate the dose consequences to both offsite and onsite receptors.

The unmitigated dose consequences of the bounding gaseous release do not exceed the offsite release limit, so no safety-class features are required to mitigate or prevent this accident. However, the dose consequences exceed the onsite risk evaluation guideline so safety-significant features have been identified to mitigate the consequences of a gaseous release. The process bay local exhaust HVAC system (for the process bays) and the process general exhaust HVAC system (for the PWC tank room) are credited with HEPA filtration of the gaseous release scenarios. In addition, the reference air system is credited with monitoring the differential pressure in the process bays to ensure that any releases are drawn through the HEPA filters. The HEPA filters remove entrained contaminated particulate material from the released gas, mitigating the DBA. With safety-significant features credited, the mitigated onsite dose is well below onsite risk evaluation guidelines. Details of the calculations for this DBA are provided in SNF-2770, Chapter 20.
B3 4 2 1 1 Scenario Development  The CVDF hazard analysis (HNF-SD-SNF-HIE-004) identified and categorized (i.e., binned) a series of potential accidents in the gaseous release accident category. This accident category includes releases from the MCO and CVDF processing equipment during various phases of the cask venting, draining, and vacuum drying process. Gaseous release accidents include leaks in a process line and subsequent removal of entrained particulate in the helium purge flow. This set of accidents involves a slightly pressurized MCO exhausting, ultimately, to the environment. This condition could exist any time there is a purge flow through the MCO.

A purge flow through the MCO could be present at any time following draining of bulk water. Initially, the purge flow will be from the filtered process exit port side to the long axially process tube side immediately following the drain and during purge and flush of the process lines. At this time a leak could occur in the PWC room. Following purge and flush, the normal purge flow will be established from the long axially process tube side to the filtered process exit port side. Following establishment of the normal purge flow through the MCO, a purge could exist, as needed, at any time prior to the end of the process. During any of these times a leak could exist in the bay.

A leak in the MCO purge line (VPS-*02-SS-1", VPS-*02-SS-2") or the connections to the MCO during helium purge will result in a gaseous release accident. The leak allows a continuous release of pressurized helium carrying particulate that could be released to the process bay. Pressure is maintained in the system because the helium lost through the leak is made up by the helium supply system. During this phase of the cold vacuum drying process, an additional 10 kg of particulate could be generated, which means that the MCO could now contain a maximum of 25 kg of particulate matter. A leak at this time is selected as the bounding and representative accident. The scenario is based on the following assumptions:

- The operating parameters are within the normal range, and the instrument and control systems do not indicate that a problem exists.
- The time period associated with helium purge following draining is nominally 0.5 hours, and the time associated with the purge following the vacuum process is nominally 4 hours. However, for the purposes of this analysis, the purge gas will be assumed to flow throughout the entire exposure period for both onsite and offsite receptors (12 and 24 hours, respectively).
- The SCIC system includes a high-pressure alarm and a low-pressure alarm for the VPS. As long as the alarm setpoints are not violated, the helium system will attempt to maintain the helium pressure by increasing purge flow to the MCO.

B3 4 2 1 2 Source Term Analysis  Section 5.2.4 of DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions/Rates for Nonreactor Nuclear Facilities*, states that the value for the aerodynamic entrainment of uranium “powders lying on a heterogeneous surface under debris or for static conditions within facilities” is $4.0 \times 10^{-6}$ per hour.
with an RF of 1.0. Before the fuel is loaded into an MCO, it is washed to remove debris and oxide. The material produced since washing and available for transport from the MCO by purge gas flow is assumed to be comparable to that of powder lying on a heterogeneous surface under debris. Thus, an ARR of $4.0 \times 10^{-6}$ per hour with an RF of 1.0 is used in this analysis. This scenario was not modeled as a pressurized release even though the pressure in the MCO at the time of the accident is above atmospheric. This is because the releasable material is partially located on the bottom of the MCO and is exposed to relatively low air velocities. The loaded internal containers in the MCO will provide substantial protection from air velocity effects produced by the purge gas flow. Thus, the conditions the releasable powder material is exposed to are more closely approximated by ambient atmospheric wind conditions than by a pressurized powder release.

Because the representative accident could happen at the end of processing the MCO at the CVDF, the total 25 kg of particulate is used in this analysis. The 25 kg of uranium dioxide particulate contains approximately 22 kg of uranium, which is the amount of the MAR for this accident. The amount of respirable material released from the MCO is calculated using the following formula

$$M = \text{MAR} \times \text{ARR} \times \text{RF} \times T$$

where

- **MAR** = material at risk (22 kg)
- **ARR** = airborne release rate ($4.0 \times 10^{-6}$ per hour)
- **RF** = respirable fraction (1.0)
- **T** = time (hours)

For a 12-hour release, the amount of respirable radionuclide material released is

$$M = (22 \text{ kg})(1,000 \text{ g/kg})(4.0 \times 10^{-6}/\text{h})(1.0)(12 \text{ h})$$

$$= 1.1 \text{ g}$$

For a 24-hour release, the amount of respirable radionuclide material released is two times the 12-hour release or $2.2 \text{ g}$

**B3 4 2 1 3 Consequence Analysis** The effective dose equivalent to a receptor is calculated by using the following equation

$$DE = M \times \frac{X}{Q'} \times BR \times UD$$

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where

\[ \text{DE} = \text{effective dose equivalent based on inhalation exposure only (rem)} \]
\[ M = \text{respirable quantity released into the air (grams)} \]
\[ \chi/Q' = \text{air transport factor (s/m³)} \]
\[ \text{BR} = \text{average inhalation rate during the release (m³/s)} \]
\[ \text{UD} = \text{committed effective dose equivalent per unit gram inhaled} \]

For the onsite receptor, the duration of worker exposure can be assumed to be 12 hours on the basis of anticipated 12-hour work shifts. For the offsite receptor, the duration of exposure is taken to be 24 hours (HNF-SD-SNF-TI-059). The air transport factor \( \chi/Q' \) for a 12-hour or a 24-hour release is the logarithmic interpolation between the 2-hour \( \chi/Q' \) with plume meander and the chronic \( \chi/Q' \) (HNF-SD-SNF-TI-059).

The light activity breathing rate of \( 3.33 \times 10^4 \text{ m}^3/\text{s} \), as defined in HNF-SD-SNF-TI-059, *A Discussion on the Methodology for Calculating Radiological and Toxicological Consequences for the Spent Nuclear Fuel Project at the Hanford Site*, is used for the onsite receptor. For the offsite receptor, an average 24-hour breathing rate of \( 2.64 \times 10^4 \text{ m}^3/\text{s} \) is used.

The dose per unit of respirable material inhaled is \( 4.38 \times 10^5 \text{ rem/g} \) of fuel, as specified in Section 3 4 1 2 of the SNF Project FSAR. All the material released from the building is treated as respirable (i.e., less than 10-μm aerodynamic diameter).

Using the respirable radionuclide release quantities calculated in Section B3 4 2 1 2, doses to various receptors are calculated as a function of leak duration.

**Unmitigated Consequences** The following dose calculation equation is used to calculate the dose to the onsite receptor:

\[
\text{DE}_{\text{onsite}} = M \times \frac{\chi}{Q'} \times \text{BR} \times \text{UD}
\]

\[
= (1 \text{ g})(6.28 \times 10^{-3} \text{ s/m}^3)(3.33 \times 10^{-4} \text{ m}^3/\text{s})(4.38 \times 10^5 \text{ rem/g})
\]

\[
= 1.0 \text{ rem} (1.0 \times 10^{-2} \text{ Sv})
\]

The unmitigated dose consequences at the remaining receptor sites are calculated in the same manner and are shown in Table B3-8.
<table>
<thead>
<tr>
<th>Receptor location</th>
<th>Duration (hours)</th>
<th>$\chi/Q^\ast$ (s/m$^3$)</th>
<th>Unmitigated dose$^b$ (rem)</th>
<th>Unmitigated evaluation guideline/release limit$^c$ (rem)</th>
<th>Mitigated dose (Sv)</th>
<th>Mitigated evaluation guideline/release limit$^d$ (rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Onsite (100 m E)</td>
<td>12</td>
<td>6.28 E-03</td>
<td>10 (10 E 02)</td>
<td>10 (10 E 02)</td>
<td>0.001 (10 E-05)</td>
<td>25 (0.25)</td>
</tr>
<tr>
<td>Columbia River (650 m W)</td>
<td>12</td>
<td>1.99 E-04</td>
<td>0.032 (3.2 E 04)</td>
<td>3.2 E-05 (3.2 E-07)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>100 Area Fire Station (3,750 m ESE)</td>
<td>12</td>
<td>2.73 E-05</td>
<td>4.4 E-03 (4.4 E-05)</td>
<td>-</td>
<td>4.4 E-06 (4.4 E-08)</td>
<td></td>
</tr>
<tr>
<td>Hanford Site boundary (10,090 m W)</td>
<td>24</td>
<td>6.50 E-06</td>
<td>1.7 E-03 (1.7 E-05)</td>
<td>0.5 (5.0 E-03)</td>
<td>1.7 E-06 (1.7 E-08)</td>
<td>5 (0.05)</td>
</tr>
</tbody>
</table>


$^\ast$Fifty year committed effective dose equivalent.

$^b$Evaluation guideline (for onsite receptor) and release limit (for offsite receptor) for unmitigated accidents in the anticipated frequency category (i.e., frequency $>10^{-5}$ per year)

$^c$Evaluation guideline/release limit based on a mitigated accident frequency of extremely unlikely
Mitigated Consequences  Since the unmitigated accident does not exceed offsite release limits, no mitigated offsite doses were calculated. For the mitigated onsite dose, the release is assumed to continue for the same period of time, but the process bay local exhaust HVAC system (including standby power to this system) for the process bays is credited with confinement of the release by filtering the release via the HEPA filters in the system. The general exhaust HVAC system is credited with filtration of releases into the process bays or process water tank room, and the differential pressure instrumentation and alarms provide indication if the general exhaust is not providing its filtering function. The mitigated onsite dose calculation includes the effects of the leak path factor for radioactive material released from the building via the HEPA filters. For a HEPA filter efficiency of 99.9%, this leak path factor is 0.001 and is applied as indicated in the following equation:

\[
DE_{\text{on-site}} = M \times \frac{X}{Q'} \times BR \times UD \times LPF
\]

\[
= (1.1 \text{ g})(6.28 \times 10^3 \text{ s/m}^3)(3.33 \times 10^4 \text{ m}^3/\text{s})(4.38 \times 10^5 \text{ rem/g})(0.001)
\]

\[
= 0.001 \text{ rem (1.0 } \times 10^5 \text{ Sv)}
\]

With safety-significant functions credited, mitigated onsite consequences for the event are 0.001 rem (1.0 \times 10^5 Sv). The mitigated gaseous release consequences also are summarized in Table B3-8.

B3.4.2.1.4 Comparison with Guidelines

Comparison of Unmitigated Doses  The unmitigated frequency for gaseous release is in the anticipated category (i.e., an unmitigated frequency greater than 1\times10^{-2} \text{ per year}) (see SNF-2770, Appendix A). The unmitigated sequence considered the processing of 200 MCOs per year and a leak from either the flexible hose connections themselves or from installation error, the piping, or the valves. The unmitigated radiological offsite dose for this event is below offsite release limits, while the unmitigated onsite dose for an anticipated event is above onsite risk evaluation guidelines.

Comparison of Mitigated Doses  The mitigated frequency of the event tree sequences that represent this DBA as an unfiltered release is 1 \times 10^{-5} \text{ per year}, thus this mitigated DBA is extremely unlikely (see SNF-2770, Appendix A). The mitigated sequence credited leak verification, local exhaust fan operation, loss of differential pressure in the process bays, response to loss of differential pressure, and HEPA filter functionality. With safety-significant features credited, the mitigated onsite dose for an extremely unlikely event is well below onsite risk evaluation guidelines.

B3.4.2.1.5 Summary of Safety-Class Structures, Systems, and Components and Technical Safety Requirement Controls  Under normal operating conditions, no unfiltered gaseous releases to the process bay or PWC tank room are expected. Under upset or accident
conditions, safety-significant equipment is required in order to ensure dose consequences do not exceed onsite risk evaluation guidelines.

The checklist designators included in the accident bins, other than the accident selected as the DBA, represent additional accident sequences slightly different than the DBA. All of these binned accidents are bounded by the DBA because they have lesser or equivalent worst-case consequences and frequencies.

In addition to the DBA, accident scenarios in the G1 bin were identified in the hazard analysis, including random process line failure, line failure due to crane load drop, local or general exhaust HEPA filter failure, random ventilation system failure (differential pressure and filtration), and PWC drain line failure (process bay, spare bay, or PWC room) (HNF-SD-SNF-HIE-004). Because of the length of piping and numerous locations where a leak could occur (e.g., fittings, valves), the general approach taken for G1 bin gaseous releases is to mitigate a release. Confinement of the release is ensured by active ventilation and HEPA filtration. To support the confinement function, the differential pressure in each bay is monitored, and the standby power system (and selected isolation dampers) ensures maintenance of confinement under a loss of facility power. However, a leak in the PWC drain line is prevented rather than mitigated because that line travels through all bays and the spare bay before entering the PWC room. Confinement provided by the ventilation systems is not maintained in the spare bay and cannot be ensured in all process bays because one of the bays could have a rollup door open.

The safety-significant equipment designated to mitigate the dose consequences of the gaseous release accident analyzed in this section is described below.

Safety-class equipment (performing a safety-significant function) for confinement

- Cask-MCO

The cask-MCO is a major part of the pressure boundary for confinement of radioactive materials during processing, its integrity prevents a gaseous release.

Safety-significant equipment for confinement and filtration

- Process bay local exhaust HVAC and process vent system (exhaust fans and plenums duct work, HEPA filters)

The process bay local exhaust HVAC and process vent system mitigates a gaseous release into the process bay by sweeping it through HEPA filters before it is discharged outside the facility.

- Process bay local exhaust HVAC and process vent system process hood isolation damper and instrument air supply.
Isolation dampers in the process bay local exhaust HVAC and process vent system process hood fail closed to ensure confinement by isolation of the unfiltered process hood. If power is lost, dampers will open with electrical power from the standby power system and instrument air supplied by the local dedicated tank. The hood isolation damper and instrument air supply operate in conjunction with the standby power system to facilitate HVAC operating while on standby power.

- Process general supply/exhaust HVAC system (exhaust HEPA filter, exhaust duct work, isolation damper)

The process general supply/exhaust HVAC system mitigates a release into the process bay or process water tank room by filtering it before discharging it outside the facility. This ensures that leaks during processing or following bulk water drain (including PWC room) are captured. The process general supply/exhaust HVAC system also provides confinement in conjunction with the facility's structure by maintaining a negative building pressure. Fail-closed exhaust dampers from the process bays and process water tank room isolate other flow paths and ensure that differential pressure is maintained.

- Process bay recirculation HVAC system isolation dampers (outside air inlets)

The process bay recirculation HVAC system provides fail-closed outside air inlet dampers so the local exhaust on standby power can maintain process bay differential pressure.

- Reference air system (reference air header, differential pressure alarms for both process bays and process water tank room)

The reference air system monitors the negative pressure in the process bays and process water tank room by providing differential pressure indication and alarms to the control room for operator response.

- PWC line between the MCO and the process water tank room

The design and installation of the PWC transfer line provide confinement of contaminated process water and gases during transfer from the MCO to the PWC room.

- Standby electrical power (diesel generator and process bay local exhaust HVAC and process vent system restart circuit)
The standby power system provides connections to restart the local exhaust fans and
supporting equipment. Operation of the local exhaust on standby power will maintain
building differential pressure sufficient for confinement during facility power outages.
Assumptions made that require protection by TSRs are listed below:

- Perform process connector leak test before processing

Before processing, a leak test must be performed to ensure that the process
connectors do not leak. This test protects safety analysis assumptions relative to air
inleakage into the MCO.

The bounding accident (G 1) and the other accidents identified in the CVDF hazard analysis
report (HNF-SD-SNF-HIE-004) that may potentially lead to a gaseous release are listed in
Table B3-9 along with corresponding checklist designators from the hazard analysis report.

The accidents in the remaining bins require the following safety SSCs and TSRs, in addition
to those identified for the DBA (G 1):

G 2 Gaseous release due to delays in shipping from the CVDF

The accident scenario identified in the hazard analysis for the G 2 bin represents a slow
release from an MCO awaiting shipment from the CVDF (HNF-SD-SNF-HIE-004). The MCO is
slightly pressurized by the fuel corrosion reaction and the MCO is leaking or a port valve was not
properly closed. The control in the G2 bin prevents the gaseous release by preventing
pressurization of the MCO that results from excessive free water remaining in the MCO.

Assumptions made that require protection by TSRs are listed below:

- Proof-of-dryness demonstration at the CVDF to minimize hydrogen generation within
  the MCO during transport to and storage at the CSB

A proof-of-dryness demonstration must be successfully completed before the final
pressure rebound test steps can begin.

G 3 Gaseous release due to line break caused by a seismic event

The accident scenario identified in the hazard analysis for the G 3 bin represents releases into
the transfer corridor or process bay following a seismic event (HNF-SD-SNF-HIE-004). The
seismic event is an unlikely event for which the onsite risk evaluation guideline is 10 rem.
Because the worst-case gaseous release dose is less than 10 rem, the guideline is satisfied without
controls.

No additional requirements result from analysis of this accident.
<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designtor</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4 0)</th>
<th>NRC ITS</th>
</tr>
</thead>
<tbody>
<tr>
<td>G 1 Gaseous release due to process line failure or HVAC failure (bounding accident, PB H 06f)</td>
<td>PB B 13a PB F 05 PB H-06d PB H 06f PB H-06g PB H 06h PB N 01 SB N-01 PW H 06</td>
<td>Prevent/mitigate gaseous release outside the facility</td>
<td>Safety-class equipment (performing a safety significant function) for confinement Cask-MCO Safety significant equipment for confinement and filtration HVAC/PV system (exhaust fans and plenums duct work, HEPA filters) HVAC/PV process hood isolation damper HVACD system (exhaust HEPA filter exhaust duct work, isolation dampers) HVACB isolation dampers (outside air inlets) PW C line between the MCO and the process water tank room Standby electrical power (diesel generator and HVAC/PV system restart circuit) Safety significant equipment for monitoring Reference air system (reference air header differential pressure alarms for both process bays and process water tank room) TSR Perform process connector leak test before processing Defense in depth Continuous air monitors</td>
<td>A B B</td>
</tr>
<tr>
<td>Candidate accident</td>
<td>Checklist designtor</td>
<td>Safety function</td>
<td>Safety features (described in Chapter B4.0)</td>
<td>NRC ITS³</td>
</tr>
<tr>
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</tr>
<tr>
<td>G 2 Gaseous release due to delays in shipping from the CVDF (slow leak)</td>
<td>PB H 061</td>
<td>Prevent out of specification MCO</td>
<td>Safety class equipment (performing a safety significant function) for confinement Cask-MCO</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Safety significant equipment for confinement and filtration HVAC/PV system (exhaust fans and plenums duct work, HEPA filters) HVAC/PV process hood isolation damper</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• HVACD system (exhaust HEPA filter exhaust duct work, isolation dampers)</td>
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<td></td>
<td></td>
<td></td>
<td>• HVACB isolation dampers (outside air inlets)</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>Standby electrical power (diesel generator and HVAC/PV system restart circuit)</td>
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<td></td>
<td></td>
<td></td>
<td>Safety significant equipment for monitoring Reference air system (reference air header differential pressure alarms for both process bays and process water tank room)</td>
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<tr>
<td></td>
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<td></td>
<td>TSR Proof of dryness demonstration at the CVDF to minimize hydrogen generation within the MCO during transport to and storage at the CSB</td>
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<td>A</td>
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<td></td>
<td></td>
<td></td>
<td>B</td>
</tr>
<tr>
<td>Candidate accident</td>
<td>Checklist designator*</td>
<td>Safety function</td>
<td>Safety features (described in Chapter B4 0)</td>
<td>NRC ITS*</td>
</tr>
<tr>
<td>--------------------</td>
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<td>---------------------------------------------------------------------------------------------------------</td>
<td>----------</td>
</tr>
</tbody>
</table>
| G 3 Gaseous release due to line break caused by a seismic event | TC R 01, PB R 01a, OU R 01a | Mitigate gaseous release These accidents meet guidelines for the unlikely event | Defense in-depth equipment for confinement and filtration  
HVACC/PV system (exhaust fans and plenums, duct work, HEPA filters)  
HVACC/PV process hood isolation damper  
HVACC system (exhaust HEPA filter, exhaust duct work, isolation dampers)  
HVACB isolation dampers (outside air inlets)  
Standby electrical power (diesel generator and HVAC/PV system restart circuit) |          |
| G 4 Gaseous release caused by facility fire | PB L 01, PB L 02, PB L 03, PB L-04, PB L-05, PB L 06, PB L-07, PB L 08, PB L 09, PB L 10, PB L 13, PB L 14, PB L 15, PB L 16, PB P-02, OU P 02a | Protect a processing bay against external fire (administrative area, transfer corridor other nonprocessing bay) and limit the fire risk inside a processing process bay | No SSCs required |
Table B3-9  Summary of Safety Features Required to Mitigate or Prevent the Consequences of a Gaseous Release  (4 sheets)

| Candidate accident | Checklist designator* | Safety function | Safety features  
(described in Chapter B4 0) | NRC ITS* |
|---------------------|-----------------------|-----------------|-----------------------------|----------|

*Checklist designator are from HNF SD SNF HIE 004 1999 Cold Vacuum Drying Facility Hazard Analysis Rev 4 Fluor Daniel Hanford, Incorporated, Richland Washington.

U.S. Nuclear Regulatory Commission important to safety classifications  Category A = critical to safe operation, Category B = major impact on safety and Category C = minor impact to safety

CSB = Canister Storage Building
CVDF = Cold Vacuum Drying Facility
HEPA = high-efficiency particulate air (filter)
HVAC = heating, ventilation, and air conditioning
HVACB = process bay recirculation HVAC system
HVACC/PV = process bay local exhaust HVAC and process vent system
HVACD = process general supply/exhaust HVAC system
ITS = important to safety
MCO = multi-canister overpack
NRC = U.S. Nuclear Regulatory Commission
PWC = process water conditioning
TSR = technical safety requirement
G 4  Gaseous release caused by facility fire

The accident scenario identified in the hazard analysis for the G 4 bin represents releases into the process bay caused by a facility fire (HNF-SD-SNF-HIE-004). The unmitigated frequency for the events in bin G 4 is unlikely (HNF-SD-SNF-HIE-004) for which the onsite risk evaluation guideline is 10 rem. Since the worst-case gaseous release dose is less than 10 rem, the guideline is satisfied without controls.

No additional requirements result from analysis of this accident.

B3 4 2 2 Liquid Release  This section addresses the consequences of a pressurized release of water and entrained contaminated particulate through a leak in the PWC system. The quantity of radioactive material released is calculated and that value is used to calculate the dose consequences to both offsite and onsite receptors.

The unmitigated dose consequences of the bounding liquid release do not exceed the offsite release limit, so no safety-class features are required to mitigate or prevent this accident. However, the dose consequences exceed the onsite risk evaluation guideline, so safety-significant features have been identified to mitigate the consequences of a liquid release. The building process general supply/exhaust HVAC system is credited with filtering the release through the HEPA filters in the system. A safety-significant differential pressure alarm is provided to detect low differential pressure to provide an indication if the general exhaust system is not providing its filtering function. With safety-significant features credited, the mitigated onsite dose is well below onsite risk evaluation guidelines. Details of the calculations for this DBA are provided in SNF-2770, Chapter 3.

B3 4 2 2 1 Scenario Development  The cask-MCO arrives at the CVDF from the K Basins with the MCO containing enough K Basin water to cover the fuel. At the CVDF, the MCO is connected to process equipment and the water is drained from the MCO. A leak in the pressurized portion of this process equipment in the process water tank room during draining could result in the release of MCO water, along with entrained particulate matter from the MCO to the process water tank room and eventually to the environment.

The CVDF hazard analysis (HNF-SD-SNF-HIE-004) identified and categorized a series of potential accidents as liquid release accidents. This accident category consists of releases from leaks developing in the PWC system. The PWC system includes the piping and other hardware involved in removing MCO water from the MCO, filtering the water, and ultimately sending the water to a holding tank until it is processed by the K Basins integrated water treatment system. The bounding event is a leak in the pressurized PWC system piping. Other locations for this accident were considered in the hazard analysis. However, the consequences were determined to be bounded by a leak that occurs at the pump outlet where the PWC system pressure is the greatest. The leak sprays water containing particulate into the process water tank room, fills the room with respirable aerosols, and creates a pool of contaminated water on the floor of the room. Additional particulate is released to the room through aerodynamic resuspension and evaporation.
from the pool’s surface. During the MCO water draining operation, the water entrains radioactive particulate in the MCO. The particulate released during the spray is ultimately released into the environment.

**B3 4 2 2 2 Source Term Analysis** The following assumptions are used in the analysis:

- The total quantity of water available for release into the process water tank room is 650 L. At the start of processing, there is 500 L in the MCO and 150 L in the receiver tanks (PWC-TK-4032, PWC-TK-4033). The water in the receiver tanks is used to prime the pump (PWC-P4035, PWC-P-4036, only one operating) in the PWC system and provide recirculation through the ejector (PWC-EJR-4031) to establish a vacuum on the drain header (PWC-001-SS-1")

- The density of the dry aerosol particulate is assumed to be 5.0 kg/L (HNF-SP-1201). The maximum density of this particulate is assumed to be 10.0 kg/L (HNF-1523).

- The spray release develops during pumping of the process water in the line from (PWC-005-SS-1") the PWC circulation water pumps. This is the point in the system with the highest pressure 60 lb/in² gauge. The liquid temperature is assumed to be 46 °C. The viscosity of water at this temperature is 0.588 centipoise.

- The inventory at risk includes the contents of the process water receiver tanks (650 L of water) and 10% of the particulate from a single MCO (15 kg) (SNF-2770). The estimate of 10% is supported by prototype tests using cerium oxide powder (HNF-4057, Section 3.4). The density of cerium oxide is 6.9 kg/L, which is high enough to represent uranium oxide. In addition, the cerium oxide particle diameters were distributed in the range from 0 μm to 10 μm, consistent with SNF particulate. Residual particulate in the PWC tanks is not included in the release calculation because the system purifies the water from previous MCO drain operations and any residual would add a negligible amount to the postulated suspended solids concentration.

- During the MCO draining operation, the drained liquid circulates around a loop that includes a pump, two tanks, and an ejector. Following the draining, the PWC system is reconfigured to circulate the contaminated MCO drain water through ion exchange modules (PWC-IXM-4037, PWC-IXM-4038, one is standby). Eventually the de-ionized liquid is transferred through a filter (PWC-F-4042) to a holding tank (PWC-TK-4001). There is no limit on the established length of time that the water can be recirculated. Therefore this accident is evaluated for a spray leak continuing for both 12 and 24 hours. These are bounding exposure times for onsite and offsite receptors downwind (HNF-SD-SNF-TI-059).
During the course of a spray release accident, a large fraction of the sprayed water falls to the floor and collects in a pool. This pool contributes a relatively small amount to the released source term. The spray release accident source term has three components included in the dose calculations:

- Particulate released with the spray of water
- Particulate resuspended from the surface of the pool of water
- Particulate released by evaporation from the pool of water

**Spray of Water** Some fraction of the solution issuing from a leak will atomize, carrying suspended SNF particulate with it. The droplets quickly evaporate leaving behind SNF particles with some water still trapped in the pores. Large leaks in piping lead to less atomization while small holes lead to lower leak rates. The SPRAY computer program (WHC-SD-GN-SWD-20007) has been used to find the bounding case hole diameter with the largest respirable release rate. User input to the program includes the water pressure, PWC liquid density and viscosity, and the largest droplet diameter that evaporates to a respirable-sized particle.

The SPRAY computer program describes the atomization of a liquid jet due to the kinetic energy of the jet itself. Because atomization of a liquid jet is a random process, the resultant spray consists of a wide range of drop sizes and must be represented by a distribution rather than a single parameter. The SPRAY computer program computes the fraction of droplets that are below a limiting diameter input by the user. It also varies the leak size to find the leak with the greatest respirable leak rate. Documentation for the SPRAY computer program (WHC-SD-GN-SWD-20007) includes user guide, verification tests, and configuration control. The code uses empirical relationships and has been validated by hand calculations.

The water pressure is assumed to be 60 lb/in², the maximum pressure expected from the PWC pump. The PWC liquid viscosity is that of water at 46 °C, namely, 0.588 centipoise. The density of the water in the PWC system is not appreciably increased by the 1.5 kg of suspended particulate.

The largest droplet diameter that will evaporate down to respirable-sized particulate is calculated to be 55 μm (SNF-2770). Airborne particles are respirable if their "aerodynamic diameter" is less than 10 μm. (The term "aerodynamic diameter" is defined to be the diameter of a unit-density sphere with the same settling velocity.) The largest SNF particle diameter that could be considered respirable is shown to be 4.3 μm. With 1.5 kg of particulate suspended in 650 L of water, the bounding liquid droplet that evaporates down to this diameter is shown to be 55 μm.

The values were input to the SPRAY code to obtain the following bounding release fractions (SNF-2770):

- The optimum hole diameter is 642 μm.
The bounding leak rate from this hole is $4.54 \times 10^{-6}$ m$^3$/s, of which $9.23 \times 10^{-8}$ m$^3$/s is respirable.

Note that these leak rates correspond to 16.3 L/h (total) and 0.332 L/h (respirable). The total leak rate determines the growth of the pool of water on the floor of the PWC tank room. The respirable leak rate determines the amount that becomes airborne from the spray of liquid from the leak.

The total volume of liquid released from the PWC system, as well as the respirable mass that becomes airborne from the spray leak, depends on the duration of the leak. Equations to represent these quantities are shown below:

$$M_{\text{spray}} = (L_{\text{spray}})(T_{\text{leak}})(M_{\text{SNF}})/(V_{\text{tot}})$$

$$V_{\text{leak}} = (L_{\text{leak}})(T_{\text{leak}})$$

where

- $M_{\text{spray}}$ = mass of SNF airborne as respirable particles by atomization during the leak (g)
- $L_{\text{spray}}$ = leak rate for respirable particles computed by the SPRAY code (0.332 L/h)
- $T_{\text{leak}}$ = duration of the leak (hours)
- $M_{\text{SNF}}$ = total mass of SNF in the PWC system (1,320 g fuel)
- $V_{\text{tot}}$ = total volume of liquid in the PWC system (650 L)
- $V_{\text{leak}}$ = volume of PWC solution that has leaked from the PWC system (L)
- $L_{\text{leak}}$ = total leak rate computed by the SPRAY code (16.3 L/h)

Note that the mass of SNF (1,320 g) is used rather than the mass of particulate (1,500 g) to calculate the mass airborne. Note also that the leak must continue for at least 40 hours for the PWC system to lose its entire inventory of liquid (650 L). The bounding exposure times at downwind receptor locations (i.e., 12 hours onsite and 24 hours offsite) apply to this release. The environmental release may be terminated after 24 hours through actions such as pump shutdown and spill cleanup.

The mass of SNF released by atomization as respirable particles after 12 hours is calculated as follows:

$$M_{\text{spray}} = (0.332 \text{ L/h})(12 \text{ h})(1,320 \text{ g})/(650 \text{ L}) = 8.09 \text{ g SNF}$$

The mass of SNF released by atomization as respirable particles after 24 hours is calculated as follows:

$$M_{\text{spray}} = (0.332 \text{ L/h})(24 \text{ h})(1,320 \text{ g})/(650 \text{ L}) = 16.18 \text{ g SNF}$$

The summary of masses released is given in Table B3-10.
Table B3-10 Quantity of Respirable Radioactive Material Released by Spray Components

<table>
<thead>
<tr>
<th>Release Component</th>
<th>Release duration</th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>12 hours</td>
<td>24 hours</td>
</tr>
<tr>
<td>Atomization by the spray</td>
<td>8.09 g</td>
<td>16.18 g</td>
<td></td>
</tr>
<tr>
<td>Resuspension from pool</td>
<td>0.0010 g</td>
<td>0.0038 g</td>
<td></td>
</tr>
<tr>
<td>Evaporation from pool</td>
<td>0.0006 g</td>
<td>0.0025 g</td>
<td></td>
</tr>
<tr>
<td>Sum of resuspension and evaporation</td>
<td>0.0016 g</td>
<td>0.0063 g</td>
<td></td>
</tr>
</tbody>
</table>

The mass released is grams of spent nuclear fuel (metal) rather than grams of particulate (uranium oxide).

**Release by Resuspension from the Pool** With respect to resuspension, DOE-HDBK-3010-94, Section 3.1, suggests that for aerodynamic entrainment and resuspension indoors, on a heterogeneous surface (stainless steel, concrete), and with low air speeds from normal facility ventilation flows, a bounding ARR is $4.0 \times 10^7$/h. Since the pool from which resuspension takes place is growing linearly with time, the total resuspended increases with the square of the time, as shown below. The factor of $\frac{1}{2}$ comes from the time integration. Since the spray leak pool volume increases linearly with time, the surface area is assumed to increase linearly with time also. This is a bounding assumption because the pool might encounter boundaries that prevent increases in surface area. With a bulk leak rate of 16.3 L/h, the volume of the pool will be 39.1 L after 24 hours. Assuming an average depth of 1 cm, the wet surface area is 39.1 m$^2$ (420 ft$^2$) at the end of 24 hours.

$$M_{RESUS} = \left(\frac{1}{2}\right)R_{RESUS}(LR_{LEAK})(T_{LEAK})\frac{(M_{SNF})}{(V_{TOT})}$$

where

- $M_{RESUS}$ = mass of SNF airborne as respirable particles by resuspension from the pool (g)
- $R_{RESUS}$ = respirable release rate from the surface of a pool by resuspension ($4 \times 10^7$/h)
- $LR_{LEAK}$ = total leak rate computed by the SPRAY code (16.3 L/h)
- $T_{LEAK}$ = duration of the leak (hours)
- $M_{SNF}$ = total mass of SNF in the PWC system (1,320 g fuel)
- $V_{TOT}$ = total volume of liquid in the PWC system (650 L)

The mass of SNF released by resuspension as respirable particles after 12 hours is calculated as follows

$$M_{RESUS} = \left(\frac{1}{2}\right)(4 \times 10^7/h)(16.3 \text{ L/h})(12 \text{ h})\frac{(1,320 \text{ g})}{(650 \text{ L})} = 0.0010 \text{ g SNF}$$
The mass of SNF released by resuspension as respirable particles after 24 hours is calculated as follows:

\[ M_{\text{RESUS}} = \left( \frac{1}{2} \right) \left( \frac{4 \times 10^7}{h} \right) (16.3 \, \text{L/h})(24 \, h)^2 \left( \frac{1320}{650} \, \text{g} \right) = 0.0038 \, \text{g SNF} \]

The summary of masses released is given in Table B3-10.

**Release by Evaporation from the Pool**  
With respect to evaporation, an upper bound of \( 5.3 \times 10^7 \) for the ARF of suspended particulate resulting from evaporation stresses is found in DOE-HDBK-3010-94, Section 3.2. This ARF is based on a 2-hour sampling time for air flowing at 0.5 m/s over small quantities of concentrated plutonium nitrate solution heated to 90 °C. These experimental conditions lead to greater emissions than those expected from conditions in the PWC room during the liquid release event. Since the spray leak pool volume increases linearly with time, the surface area is assumed to increase linearly with time as described earlier. This is a bounding assumption because the pool might encounter boundaries that prevent increases in surface area. The release of suspended particulate due to evaporation from this growing pool increases linearly with time. The total amount of suspended particulate released from the pool increases with the square of the time, as shown below. The factor of \( \left( \frac{1}{2} \right) \) comes from the time integration.

\[ M_{\text{EVAP}} = \left( \frac{1}{2} \right) \left( R_{\text{EVAP}} \right) \left( L_{\text{LEAK}} \right) \left( T_{\text{LEAK}} \right)^2 \left( M_{\text{SNF}} \right) / \left( V_{\text{TOT}} \right) \]

where:

- \( M_{\text{EVAP}} \) = mass of SNF airborne as respirable particles by evaporation from the pool (g)
- \( R_{\text{EVAP}} \) = average respirable release rate from the surface of a pool by evaporation \( (5.3 \times 10^7/2 \, h = 2.6 \times 10^7/\text{h}) \)
- \( L_{\text{LEAK}} \) = total leak rate computed by the SPRAY code (16.3 L/h)
- \( T_{\text{LEAK}} \) = duration of the leak (hours)
- \( M_{\text{SNF}} \) = total mass of SNF in the PWC system (1,320 g fuel)
- \( V_{\text{TOT}} \) = total volume of liquid in the PWC system (650 L)

The mass of SNF released by evaporation as respirable particles after 12 hours is calculated as follows:

\[ M_{\text{EVAP}} = \left( \frac{1}{2} \right) \left( 2.6 \times 10^7/\text{h} \right) (16.3 \, \text{L/h})(12 \, h)^2 \left( \frac{1320}{650} \, \text{g} \right) = 0.0006 \, \text{g} \]
The mass of SNF released by evaporation as respirable particles after 24 hours is calculated as follows:

\[ M_{\text{EVAP}} = \left(\frac{1}{2}\right) \left(2.6 \times 10^7 \text{h}\right) \left(16.3 \text{ L/h}\right) \left(24 \text{ h}\right)^2 \left(1.320 \text{ g}/(650 \text{ L})\right) = 0.0025 \text{ g} \]

The summary of masses released is given in Table B3-10.

From the release quantities presented in Table B3-10, it can be seen that the combined resuspension and evaporation from the pool is less than three orders of magnitude lower than the spray release. Thus, a spill of contaminated water might result in a localized spread of contamination but would not result in a significant release. If the water were filtered (e.g., a spill sometime after passing through the ion-exchange modules), the release would be even less.

**B3.4.2.2.3 Consequence Analysis** The radiological dose (effective dose equivalent) to a receptor is calculated by using the following equation:

\[ D_E = M \times \frac{x}{Q'} \times BR \times UD \]

where:
- \( D_E \) = effective dose equivalent based on inhalation exposure only (rem)
- \( M \) = respirable particulate quantity released into the air (grams)
- \( \frac{x}{Q'} \) = air transport factor (s/m³)
- \( BR \) = average inhalation rate during the release (m³/s)
- \( UD \) = committed effective dose equivalent per unit gram inhaled

For the onsite receptor, the duration of worker exposure can be assumed to be 12 hours on the basis of an anticipated 12-hour work shift. For the offsite receptor, the duration of exposure is taken to be 24 hours. For releases occurring over a 12-hour or a 24-hour time period, the air transport factor \( \frac{x}{Q'} \) is the logarithmic interpolation between the 1-hour to 2-hour \( \frac{x}{Q'} \) with plume meander and the annual average \( \frac{x}{Q'} \) (HNF-SD-SNF-TI-059). The logarithmic interpolation is a method recommended in NRC Regulatory Guide 1.145 for release durations between 2 hours and 1 year.

The light activity breathing rate of \( 3.33 \times 10^4 \text{ m}^3/\text{s} \), as defined in HNF-SD-SNF-TI-059, is used for the onsite receptor. For the offsite receptor, an average 24-hour breathing rate of \( 2.64 \times 10^4 \text{ m}^3/\text{s} \) is used.

The dose per unit of respirable material inhaled is \( 4.38 \times 10^5 \text{ rem/g} \) of fuel, as specified in Section 3.4.1.2 of the SNF Project FSAR. All the material released from the building is treated as respirable (i.e., less than 10 μm aerodynamic diameter).
Unmitigated Consequences  For the collocated worker (100 m east of the CVDF) during a 12-hour leak event, the dose from the spray atomization portion of the leak is calculated as follows

\[ D_{\text{onsite}} = M \times \frac{X}{Q'} \times BR \times UD \]

\[ = (8.09 \text{ g})(6.28 \times 10^3 \text{ s/m}^3)(3.33 \times 10^{-4} \text{ m}^3/\text{s})(4.38 \times 10^5 \text{ rem/g}) \]

\[ = 74 \text{ rem (0.074 Sv)} \]

Using the respirable radionuclide releases from Table B3-10 and the air transport factors from Table B3-11, doses at the various receptor locations as a function of the spray leak duration are calculated as demonstrated below. The results are summarized in Table B3-11.

Mitigated Consequences  The release in the mitigated case is assumed to continue for the same period of time, but the building process general supply/exhaust HVAC system is credited with confinement of the release by maintaining a negative differential pressure and filtering the release via the HEPA filters in the system. The mitigated dose calculation includes the effects of the leak path factor involved with the radioactive material released exiting the building via the HVAC system HEPA filters. If a HEPA filter efficiency of 99.9% is assumed, this leak path factor is 0.001 and is applied as indicated in the following equation:

\[ D_{\text{onsite}} = M \times \frac{X}{Q'} \times BR \times UD \times LPF \]

\[ = (8.09 \text{ g})(6.28 \times 10^3 \text{ s/m}^3)(3.33 \times 10^{-4} \text{ m}^3/\text{s})(4.38 \times 10^5 \text{ rem/g})(0.001) \]

\[ = 7.4 \times 10^3 \text{ rem (7.4 \times 10^{-1} Sv)} \]

With safety-significant functions credited, mitigated onsite consequences for the event are 7.4 \times 10^3 rem.

B3 4 2 2 4 Comparison with Guidelines

Comparison of Unmitigated Doses  Event tree sequences indicate an unmitigated frequency for the liquid release DBA is in the anticipated category (i.e., an unmitigated frequency greater than 10^{-2} per year) (see SNF-2770, Appendix A). The unmitigated sequence considered the processing of 200 MCOs per year and a leak from either the PWC pump, the piping, PWC valves, or the flexible hose connections. The unmitigated radiological offsite dose for this event is below offsite release limits, while the unmitigated onsite dose for an anticipated event is above onsite risk evaluation guidelines.

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<table>
<thead>
<tr>
<th>Receptor location (distance direction)</th>
<th>Duration (hours)</th>
<th>$\chi/Q$ * (s/m²) (without stack)</th>
<th>Unmitigated dose* (rem) (Sv)</th>
<th>Unmitigated evaluation guideline/release limit* (rem) (Sv)</th>
<th>Mitigated dose rem (Sv)</th>
<th>Mitigated evaluation guideline/release limit* (rem) (Sv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Onsite (100 m E)</td>
<td>12 (spray)</td>
<td>6.28 E-03</td>
<td>7.4 (0.074)</td>
<td>1.0 (1.0E-02)</td>
<td>7.4 E-03 (7.4 E-05)</td>
<td>25 (0.25)</td>
</tr>
<tr>
<td></td>
<td>12 (pool)</td>
<td>6.28 E-03</td>
<td>1.5 E-03 (1.5 E-05)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Total</td>
<td>6.28 E-03</td>
<td>7.4 (0.074)</td>
<td>1.0 (1.0E-02)</td>
<td>7.4 E-03 (7.4 E-05)</td>
<td>25 (0.25)</td>
</tr>
<tr>
<td>Columbia River (650 m W)</td>
<td>12 (spray)</td>
<td>1.99 E-04</td>
<td>0.23 (2.3 E-03)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>12 (pool)</td>
<td>1.99 E-04</td>
<td>4.6 E-05 (4.6 E-07)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Total</td>
<td>1.99 E-04</td>
<td>0.23 (2.3 E-03)</td>
<td></td>
<td>2.3 E-04 (2.3 E-06)</td>
<td>--</td>
</tr>
<tr>
<td>100 Area Fire Station (3,750 m ESE)</td>
<td>12 (spray)</td>
<td>2.73 E-05</td>
<td>0.032 (3.2 E-04)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>12 (pool)</td>
<td>2.73 E-05</td>
<td>6.4 E-06 (6.4 E-08)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Total</td>
<td>2.73 E-05</td>
<td>0.032 (3.2 E-04)</td>
<td></td>
<td>3.2 E-05 (3.2 E-07)</td>
<td>--</td>
</tr>
<tr>
<td>Hanford Site Boundary (10 090 m W)</td>
<td>24 (spray)</td>
<td>6.50 E-06</td>
<td>1.2 E-02 (1.2 E-04)</td>
<td></td>
<td>1.2 E-05 (1.2 E-07)</td>
<td>5 (0.05)</td>
</tr>
<tr>
<td></td>
<td>24 (pool)</td>
<td>6.50 E-06</td>
<td>4.7 E-06 (4.7 E-08)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Total</td>
<td>6.50 E-06</td>
<td>1.2 E-02 (1.2 E-04)</td>
<td>0.5 (5.0 E-03)</td>
<td>1.2 E-05 (1.2 E-07)</td>
<td>5 (0.05)</td>
</tr>
</tbody>
</table>


*Fifty year committed effective dose equivalent

*Evaluation guideline (for onsite receptor) and release limit (for offsite receptor) for unmitigated accidents in the anticipated frequency category (i.e., frequency >10⁻² per year)

*Because the unmitigated accident does not exceed offsite release limits, mitigated consequences were only calculated for onsite receptor locations.
**Comparison of Mitigated Doses** The mitigated frequency of the event tree sequences that represent this DBA as an unfiltered release is $3 \times 10^{-6}$ per year, thus this mitigated DBA is extremely unlikely (see SNF-2770, Appendix A). The mitigated sequence credited differential pressure monitoring in the PWC room, operator response to shut off the pump, and general exhaust HVAC system HEPA filter functionality. With safety-significant features credited, the mitigated onsite dose for an extremely unlikely event is well below onsite risk evaluation guidelines.

**B3 4 2 2 5 Summary of Safety-Class Structures, Systems, and Components and Technical Safety Requirement Controls** Under normal operating conditions, no liquid releases to the process bay or process water tank room are expected. Under upset or accident conditions, safety-significant equipment is required in order to ensure dose consequences do not exceed onsite risk evaluation guidelines.

The checklist designators included in the accident bins, other than the accident selected as the DBA, represent additional accident sequences slightly different than the DBA. All of these binned accidents are bounded by the DBA because they have lesser or equivalent worst-case consequences and frequencies.

In addition to the DBA, an accident was identified in the L1 bin representing a spray leak from a random line failure in the PWC room (HNF-SD-SNF-HIE-004). The controls in the L1 bin provide mitigation of the liquid release (spray) by HEPA filtration. The filtration function is monitored, and if it is lost, compensatory measures are taken to make sure that a release cannot occur.

The safety-significant equipment designated to prevent or mitigate the dose consequences of the liquid release accident analyzed in this section is described below.

Safety-significant equipment for confinement and filtration:

- **Process general supply/exhaust HVAC system (exhaust HEPA filter exhaust duct work)**

  The process general supply/exhaust HVAC system mitigates a spray leak in the process water tank room by filtering it before discharging it outside the facility. The process general supply/exhaust HVAC system also provides confinement in conjunction with the facility's structure by maintaining a negative building pressure.

- **Reference air system (reference air header, differential pressure alarms for process water tank room)**

  The reference air system monitors the negative pressure in the process water tank room by providing differential pressure indication and alarms to the control room for operator response. Such monitoring ensures that the air in the PWC room is being
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drawn into the process general supply/exhaust HVAC system so that the credited filtration function is being performed. The alarms allow operators to take action to stop PWC pump operation (see TSR below) if differential pressure is lost in the PWC room (e.g., a component failure halts ventilation flows)

Assumptions made that require TSR controls are listed below

- Stop PWC pump operation on loss of differential pressure in the PWC room

If differential pressure is lost in the PWC room, PWC pump operation must be terminated. This control ensures that a spray leak does not occur at a time when filtration by the general exhaust system is not available.

The bounding liquid release accident (L1) and the other accidents identified in the CVDF hazard analysis report (HNF-SD-SNF-HIE-004) that may potentially lead to a liquid release are listed in Table B3-12 along with corresponding checklist designators from the hazard analysis.

The accidents in the remaining bins require the following safety SSCs and TSRs, in addition to the ones identified for the DBA (L1)

L2 Spray release due to facility fire

The accident scenarios identified in the hazard analysis for the L2 bin represent a spray leak in the PWC room caused by a fire (HNF-SD-SNF-HIE-004). The control in the L2 bin precludes a liquid release (spray) by precluding the possibility of a significant fire. Although the worst-case liquid release consequences are less than the onsite risk evaluation guideline of 10 rem for the unmitigated fire (an unlikely event according to HNF-SD-SNF-HIE-004), the control is credited to preclude the possibility of multiple sprays.

Assumptions made that require protection by TSRs are listed below

- Combustible loadings limited

While an MCO is present in the facility, combustible loadings are limited as determined by the fire hazard analysis implementation plan (SNF-4942). These limits ensure that any fire in the CVDF does not result in uncontrolled releases (e.g., fire-caused loss of process control).

L3 Spray release due to line break caused by a seismic event

The accident scenario identified in the hazard analysis for the L3 bin represents a spray leak in the PWC room following a seismic event (HNF-SD-SNF-HIE-004). For this event, which is unlikely in the unmitigated case, no controls are necessary because the event meets all onsite risk evaluation guidelines and offsite release limits. The unmitigated frequency for the seismic
Table B3-12  Summary of Safety Features Required to Mitigate or Prevent a Liquid Release

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator*</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4.0)</th>
<th>NRC ITSa</th>
</tr>
</thead>
<tbody>
<tr>
<td>L 1 Spray release due to piping failure (bounding accident, PW H 06)</td>
<td>PW H 06 PW N 01</td>
<td>Prevent/mitigate release outside the facility</td>
<td>Safety significant equipment for confinement and filtration HVACD system (exhaust HEPA filter exhaust duct work) Safety significant equipment to provide monitoring Reference air system (reference air header differential pressure alarms for process water tank room) whenever the PWC pump is operating TSR Stop PWC pump operation on loss of differential pressure in PWC room Defense in depth Continuous air monitors in the room PWC room floor design to contain spills and spray Liquid level detection alarm on PWC skid</td>
<td>B</td>
</tr>
<tr>
<td>L 2 Spray release due to fire</td>
<td>PW L 02 PW L 08 PW L 14 PW L 16</td>
<td>Prevent/mitigate a fire in the process water tank room</td>
<td>TSR Combustible loadings limited Defense in depth Fire protection system</td>
<td></td>
</tr>
<tr>
<td>L 3 Spray release due to a seismic event</td>
<td>OU R 01b</td>
<td>Prevent/mitigate liquid spray release outside the facility</td>
<td>No safety SSCs required Defense in depth Process water conditioning system (seismic detector and pump trip)</td>
<td></td>
</tr>
</tbody>
</table>

*Checklist designators are from HNF SD SNF HIE-004 1999 Cold Vacuum Drying Facility Hazard Analysis Rev 4 Fluor Daniel Hanford

Incorporated Richland Washington

*U S Nuclear Regulatory Commission important to safety classifications Category A = critical to safe operation, Category B = major impact on safety

Category C = minor impact to safety

*No emergency power is needed to power the fans. In case of loss of power the PWC process will be stopped

HEPA = high efficiency particulate air (filter)  HVACD = process general supply/exhaust HVAC system  ITS = important to safety  NRC = U S Nuclear Regulatory Commission  PWC = process water conditioning  SSC = structure system, and component  TSR = technical safety requirement
event is unlikely (HNF-SD-SNF-HIE-004), for which the onsite risk evaluation guideline is 10 rem. Because the worst-case liquid release is less than 10 rem, the guideline is satisfied without controls. In addition, a single optimized spray leak bounds multiple non-optimized leaks because the dose consequences of a spray release drop rapidly as the leak departs from optimal conditions.

Defense in depth

- Process water conditioning system (seismic detector and pump interlock)

The PWC system’s seismic detector provides a signal to the PWC pump shutdown interlock to stop the process water pumps during a seismic event.

B3 4 2 3 Multi-Canister Overpack External Hydrogen Explosion This section addresses the consequences of an MCO external hydrogen explosion. The quantity of radioactive material released is calculated and that value is used to calculate the dose consequences to both offsite and onsite receptors.

The unmitigated dose consequences of the bounding MCO external hydrogen explosion do not exceed the offsite release limit, so no safety-class features are required to mitigate or prevent this accident. However, the dose consequences exceed the onsite risk evaluation guidelines, so safety-significant features have been identified to mitigate the consequences of an external hydrogen explosion. Safety-significant features selected to prevent buildup of flammable concentrations of hydrogen for the bounding external hydrogen explosion DBA include a cask vent jumper tool, MCO vent jumper, a fixed orifice in the cask vent line to restrict hydrogen flow rate during cask venting, a minimum flow rate in the process bay local exhaust HVAC system, and an interlock that interrupts cask venting if a low process bay local exhaust HVAC flow rate is detected. In addition, the particulate buildup on the HVAC HEPA filters is limited by measuring the radiation level near the filter banks and changing out the HEPA filters that register above a predetermined limit. Limiting particulate buildup on the HEPA filters provides an additional barrier that mitigates the potential dose consequences from external hydrogen explosion events. With safety-significant features credited, the mitigated onsite dose is below onsite risk evaluation guidelines. Details of the calculations for this DBA are provided in SNF-2770, Chapter 4.

B3 4 2 3 1 Scenario Development When the cask-MCO first arrives at the CVDF, it contains helium and hydrogen in the void space over the water-covered fuel. The helium is introduced at K Basins to displace air from the cask-MCO headspace. Hydrogen is produced during transfer of the cask-MCO to the CVDF primarily by the corrosion of uranium and also by the radiolytic decomposition of water. One of the first actions at the CVDF is to vent the excess pressure created by hydrogen generation to the process bay local exhaust HVAC and process vent system. The cask lid can then be safely removed and replaced with the process hood. If significant hydrogen has accumulated in the cask-MCO before venting, an explosive mixture of hydrogen and air could be formed in the process bay local exhaust HVAC and process vent system when the cask-MCO is vented.
B3 4 2 3 2 Source Term Analysis  The following assumptions are used in the analysis

- The process bay local exhaust HVAC and process vent system is operating at its normal flow rate, and the pressure in the cask is relieved quickly, or the exhaust system is operating at reduced flow rates, and the pressure in the cask is relieved slowly. Either of these cases could produce flammable mixtures of hydrogen and air in the process bay local exhaust HVAC and process vent system.

- An ignition source is present near the process bay local exhaust HVAC and process vent system HEPA filter array. This source could be from rapid oxidation of uranium hydride or static electrical discharge. The resulting hydrogen deflagration causes release of the radioactive material on the filters to the environment.

- The MCO is transported on a hot day within the bounding transportation and preparation times. The bounding hydrogen generation rate can be estimated from the bounding corrosion rate estimated to occur while the fuel is wet. The hydrogen generation rate depends on the assumed fuel temperature. During shipment from the K Basins to the CVDF, the average fuel temperature is assumed to increase from 16 °C (61 °F) to 43 °C (109 °F) according to the time and temperature projections for a hot, sunny day (HNF-SD-W441-CN-001). The bounding shipping window duration is 24 hours. This is the time from cask closure at the K Basins to cask venting at the CVDF. The total hydrogen generated during the shipping window is 8.29 moles of H₂ (SNF-2770).

- The void space in the cask–MCO during shipment is estimated to be 41.2 L. This is based on a gas layer 4.25 in thick and 20 in diameter at the top of the MCO. In addition, there is a layer 0.6 in thick between the outside of the MCO and the inside of the cask. This region is treated as a cylindrical shell 16 in tall with a diameter of 24 in. The space between the bottom of the cask lid and the top of the MCO is a cylinder 1 in tall with a diameter of 24 in. All volumes are taken from the MCO Topical Report (HNF-SD-SNF-SARR-005).

- It is estimated that a HEPA filter housing that reads 100 mR/h on contact contains about 131 g of fuel (SNF-2770). This includes a factor of 10 to account for possible loss of 137Cs due to radioactive decay and greater solubility.

The mathematical analysis of the cask venting accident focuses on the hydrogen concentration in the process bay local exhaust HVAC and process vent system duct work.

The void space in the cask–MCO is filled initially with helium at a pressure of 3 lb/in² gauge (HNF-SD-SNF-OCD-001). From the ideal gas law, the cask initially contains 2.06 g moles of helium. The gas temperature is assumed to be 20 °C (68 °F) because the helium is added before the cask–MCO heats up during transport. With the addition of 8.29 moles of hydrogen...
and 0.09 gmoles water vapor, the absolute pressure in the cask at the time it is vented is 6.41 atm or 79.6 lb/in² gauge.

The gas in the cask is vented to the process bay local exhaust HVAC and process vent system at atmospheric pressure and at a temperature of 30 °C (86 °F). This temperature is several degrees below the temperatures assumed in the shipping analysis (HNF-SD-TP-SARP-017) for conditions after arrival at the CVDF. This temperature assumption has minimal effect on the calculated fraction of cask-MCO gas that leaves during venting. The amount of hydrogen vented is computed in two steps. The first step is to compute the quantity of the hydrogen-helium mixture that will leave the MCO during venting. The second step is to apply the mole ratio of hydrogen to helium to calculate just the hydrogen amount.

The volume of the hydrogen-helium gas mixture that will leave the MCO cask during venting is the total amount of gas minus the amount left in the cask at atmospheric pressure:

\[10.44 \text{ gmoles} - 1.63 \text{ gmoles} = 8.81 \text{ gmoles}\]

The second step is to calculate the amount of hydrogen that leaves the cask and enters the process bay local exhaust HVAC and process vent system. Of the 10.44 moles of gas in the MCO, 8.29 moles (79.4%) are hydrogen. Thus, the amount of hydrogen vented is calculated to be 7.00 gmoles.

To evaluate hydrogen concentrations in the process bay local exhaust HVAC and process vent system duct work, it is necessary to have values for the flow rates of both hydrogen and air into the duct. To estimate the rate at which the hydrogen in the cask would enter the local exhaust process vent system, it is assumed that the vent connection on the lid of the cask has a diameter of 0.5 in (HNF-SD-TP-SARP-017). Gases cannot exit a hole faster than the speed of sound. The minimum cask pressure needed to cause sonic flow has been calculated to be 49.2 lb/in² absolute (SNF-2770). The expected pressure of 79.6 lb/in² gauge is enough to cause sonic flow. The sonic flow rate has been computed to be 22.3 gmoles/s (0.125 lbm/s). Flow through a flexible line would reduce this value. Nevertheless, the majority of the cask hydrogen will enter the process bay local exhaust HVAC and process vent system in less than 1 second. It is therefore assumed that all 7.00 gmoles of H₂ enter the duct in 1 second. If the flow were less than sonic velocity, the hydrogen would take longer to vent, resulting in lower concentrations in the exhaust duct.

If the hydrogen in the duct were ignited near the process bay local exhaust HVAC and process vent system HEPA filter, the blast wave could damage the HEPA filters and their housing, allowing radioactive contamination on the filter to be released. The amount of radioactivity released to the environment by this hydrogen can be estimated using bounding assumptions. HEPA filters normally are changed when the differential pressure becomes too large, indicating that the filter is becoming plugged. In addition, the dose rate on the housing to the side of the filters is limited administratively. Since the filters must be changed by hand, the exposure to personnel during changeout of filters would be excessive if the dose rate limit were
not in place. For HEPA filter blasts, DOE-HDBK-3010-94, Section 5.2.2.2, has a bounding release fraction of 0.01 with an RF of 1.0. For a filter loaded with the amount of fuel (131 g) needed to give a reading of 100 mR/h, the airborne release is 1.31 g fuel.

B3 4.2.3.3 consequence analysis. The radiological dose (effective dose equivalent) to a receptor is calculated by using the following equation:

\[ DE = M \times \frac{X}{Q'} \times BR \times UD \]

where

- **DE** = effective dose equivalent based on inhalation exposure only (rem)
- **M** = respirable particulate quantity released into the air (grams)
- **\( \frac{X}{Q'} \)** = air transport factor (s/m³)
- **BR** = average inhalation rate during the release (m³/s)
- **UD** = committed effective dose equivalent per unit gram inhaled

The respirable particulate quantity is the 1.31 g fuel calculated in Section B3 4.2.3.2. The value for \( \frac{X}{Q'} \) is that shown in Table B3-13. The light activity breathing rate is \( 3.33 \times 10^{-4} \) m³/s, as defined in HNF-SD-SNF-TI-059. The dose per unit of respirable material inhaled is \( 4.38 \times 10^5 \) rem/g, as specified in Section 3.4.1.2 of the SNF Project FSAR.

**Unmitigated Consequences**. The unmitigated dose to the onsite receptor is calculated as follows:

\[ D_{\text{on site}} = M \times \frac{X}{Q'} \times BR \times UD \]

\[ = (1.31 \text{ g})(7.32 \times 10^{-2} \text{ s/m}^3)(3.33 \times 10^{-4} \text{ m}^3/\text{s})(4.38 \times 10^5 \text{ rem/g}) \]

\[ = 14 \text{ rem (0.14 Sv)} \]

The dose consequences at the remaining receptor sites are calculated in the same manner and are shown in Table B3-13.

**Mitigated Consequences**. Mitigation of the cask venting accident involves (1) preventing flammable concentrations in the process bay local exhaust duct work, and (2) ensuring the HEPA filter loading is low enough that a flammable gas explosion in the filters would not release enough activity to the environment to exceed the onsite risk evaluation guideline.
Table B3-13  Dose Calculation Summary for Hydrogen Explosion at the Process Bay Local Exhaust Heating, Ventilation, and Air Conditioning and Process Vent System Filter

<table>
<thead>
<tr>
<th>Receptor location (distance direction)</th>
<th>Duration (hours)</th>
<th>$\chi/Q$ ($s/m^3$) (without stack or plume meander)</th>
<th>Unmitigated dose(b) rem (Sv)</th>
<th>Unmitigated evaluation guideline/release limit rem (Sv)</th>
<th>Mitigated dose rem (Sv)</th>
<th>Mitigated evaluation guideline/release limit(e) rem (Sv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Onsite (100 m E)</td>
<td>&lt;1</td>
<td>7.32E-02</td>
<td>14 (0.14)</td>
<td>10 (1.0E-02)</td>
<td>10 (1.0E-02)</td>
<td>25 (0.25)</td>
</tr>
<tr>
<td>Columbia River (650 m W)</td>
<td>&lt;1</td>
<td>2.44E-03</td>
<td>0.47 (0.0047)</td>
<td>--</td>
<td>0.33 (3.3E-03)</td>
<td></td>
</tr>
<tr>
<td>100 Area Fire Station (3 750 m ESE)</td>
<td>&lt;1</td>
<td>1.60E-04</td>
<td>3.1E-02 (3.1E-04)</td>
<td>2.2E-02 (2.2E-04)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Hanford Site boundary (10 090 m W)</td>
<td>&lt;1</td>
<td>4.48E-05</td>
<td>8.6E-03 (8.6E-05)</td>
<td>6.1E-03 (6.1E-05)</td>
<td>5 (0.05)</td>
<td></td>
</tr>
</tbody>
</table>


\(a\) Fifty year committed effective dose equivalent

\(b\) Evaluation guideline (for onsite receptor) and release limit (for offsite receptor) for unmitigated accidents in the anticipated frequency category (i.e. frequency >10\(^{-2}\) per year)

\(e\) Evaluation guideline/release limit based on a mitigated accident frequency of extremely unlikely
A flammable mixture in the duct work may result from two situations (1) rapid venting of the cask into normal process bay local exhaust air flow, or (2) venting of the cask into a duct with no air flow. The first situation is prevented by the presence of a flow-reducing orifice in the vent line. The second situation is prevented by a valve that is interlocked to local exhaust flow in each bay. If local exhaust flow is lost during the venting operation, the flow of gas from the cask is halted. Both the design of the vent line and the reliability of the process bay local exhaust system ensure that there is only a low probability of a flammable mixture forming.

The activity on the process bay local exhaust filters can be limited by limiting the dose rate near the filters. Routine surveys are performed by health physics technicians to measure dose rates near the filters. The routine measurements ensure that the filter dose rates do not exceed administrative criteria chosen to limit exposure to workers who must change the filters. Normally, filters are changed if the differential pressure indicates reduced flow rates. Excessive dose rates can also trigger a filter change. Filter changes involve contact handling of the filters and may result in large personnel doses if the filter dose rates are high.

To ensure that onsite risk evaluation guidelines would not be exceeded from a hydrogen explosion involving the local exhaust HEPA filters, a limitation on the dose rate near the filter housing has been developed. A dose rate of 82 mR/h on contact at the filter housing corresponds to 94 g of SNF on the filters, assuming that the relative amounts of $^{137}$Cs and $^{241}$Am are 10 times lower on the filters than in the SNF (SNF-2770). The explosion would release 1% of the activity (bounding) on the filters, so that 0.94 g of SNF would be released as respirable particles. This release would result in an onsite dose of 10 rem, which is equal to the onsite risk evaluation guideline (unlikely frequency category). The conservative nature of the dose calculation combined with the reduced frequency of the mitigated event ensures that this dose rate limit (82 mR/h) is adequate to protect onsite workers.

**B3 4 2 3 4 Comparison to Guidelines**

**Comparison of Unmitigated Doses** Event tree sequences indicate that the unmitigated frequency for the external hydrogen explosion DBA is in the anticipated category (i.e., an unmitigated frequency greater than $10^{-2}$ per year) (see SNF-2770, Appendix A). The unmitigated sequence considered the processing of 200 MCOs per year and sufficient hydrogen accumulation in cask prior to venting. The unmitigated radiological offsite dose for this event is below offsite release limits, while the unmitigated onsite dose for an anticipated event is above onsite risk evaluation guidelines.

**Comparison of Mitigated Doses** The mitigated frequency of the event tree sequences that represent this DBA as an external hydrogen explosion and unfiltered release is $2 \times 10^{-6}$ per year thus this mitigated DBA is extremely unlikely (see SNF-2770 Appendix A). The mitigated sequence credited installation of flow restriction in the cask vent line, local exhaust running, and HEPA filter loading. With safety-significant features credited, the mitigated onsite dose for an extremely unlikely event is below onsite risk evaluation guidelines.
B3 4 2 3 5 Summary of Safety-Class Structures, Systems, and Components and Technical Safety Requirement Controls

Under normal operating conditions, there is no external accumulation of flammable concentrations of hydrogen. Under upset or accident conditions, safety-significant equipment is required in order to ensure flammable concentrations of hydrogen external to the MCO and CVDF systems are precluded.

The checklist designators included in the accident bins, other than the accident selected as the DBA, represent additional accident sequences slightly different than the DBA. All of these binned accidents are bounded by the DBA because they have lesser or equivalent worst-case consequences and frequencies.

In addition to the DBA, an accident was identified in the E1 bin representing an external hydrogen explosion caused by elevated hydrogen generations from misloaded fuel in an MCO or a shipping delay to CVDF (HNF-SD-SNF-HIE-004). The controls in the E1 bin provide both prevention and mitigation of an external explosion event, as well as protection of initial condition assumptions made in the analysis. The accident is prevented by directing the hydrogen from the cask to the local exhaust where it is diluted to less than flammable concentrations, and by terminating the vent operation if the dilution airflow is lost in the local exhaust duct. These preventative controls make the accident unlikely (SNF-2770, Appendix A) for which the onsite risk evaluation guideline is 10 rem. The dose consequence of the worst-case external hydrogen explosion is greater than 10 rem, so credit is taken for a limit on the material at risk (on the HEPA filters) to provide mitigation of the consequences. With this additional control, the frequency of the accident is reduced to extremely unlikely, and the dose consequence falls within risk evaluation guidelines. One additional control is applied to protect initial cask-MCO conditions that were assumed in the analysis.

The safety-significant equipment is described below.

Safety-significant equipment for dilution:

- Cask venting orifice and jumper

The process bay local exhaust HVAC and process vent system provides a cask venting connection with a flow restricting orifice to keep concentrations of hydrogen below the flammable limit with at least 1,000 ft³/min of flow in the duct work. The cask vent jumper tool also directs hydrogen from the cask to the local exhaust to provide for worker safety.

- MCO vent jumper tool

The MCO venting tool provides a means to vent an MCO prior to process connections being made (after the cask lid has been removed). This vent tool controls MCO pressures in case of process delays by venting to the local exhaust through the same vent line (including orifice restriction). The vent tool also provides for worker safety by directing the hydrogen from the cask to the local exhaust.
Cask venting interlock with process bay local exhaust HVAC and process vent system

A shutoff valve interlocked to the local exhaust low-flow alarm is provided because flammable concentrations would be generated almost instantaneously if the cask were vented into a stagnant local exhaust duct.

Process bay local exhaust HVAC and process vent system (exhaust fans and plenums, duct work, HEPA filters, and flow switch)

The process bay local exhaust HVAC and process vent system mitigates a gaseous release into the process bay by sweeping it through HEPA filters before it is discharged outside the facility. Low flow alarms are set at 1,150 ft³/min to provide an actual flow of greater than 1,000 ft³/min. The flow switch is interlocked with the shutoff valve.

Safety-class equipment (performing a safety-significant function) for confinement

Cask-MCO

The cask-MCO is a major part of the pressure boundary for confinement of radioactive materials during processing.

Assumptions made that require protection by TSRs are listed below

Receipt transportation window

The transportation cask must be vented to the process bay local exhaust HVAC and process vent system HEPA filters within 24 hours from being sealed at the K Basins.

Limitation on the radionuclide inventory on the process bay local exhaust HVAC and process vent system HEPA filters

The HEPA filter and prefilter housing contact dose radiation levels must be maintained equal to or less than 70 mR/h consistent with the source term used in the accident analysis.

The bounding MCO external hydrogen explosion accident (E 1) and the other accidents identified in the CVDF hazard analysis report (HNF-SD-SNF-HIE-004) that may potentially lead to a flammable hydrogen mixture outside the MCO are listed in Table B3-14, along with corresponding checklist designators from the hazard analysis report, safety functions, and SSCs.

The accidents in the remaining bins require the following safety SSCs and TSRs, in addition to the ones identified for the DBA (E 1).
## Table B3-14 Summary of Safety Features Required to Mitigate or Prevent a Multi-Canister Overpack External Hydrogen Explosion (7 sheets)

<table>
<thead>
<tr>
<th>Checklist designator</th>
<th>Candidate accident</th>
<th>Safety function</th>
<th>Safety features</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>E1: Hydrogen explosion outside an MCO, due to hydrogen generation within the cask (BBL 11a)</td>
<td>Prevent hydrogen explosion</td>
<td>Safety significant equipment for confinement and dilution</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Dilute the hydrogen in the exhaust system</td>
<td>Cask vent/jumper tool</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Direct hydrogen to the exhaust system (operator safety)</td>
<td>Cask vent/micro-cool system (HVAC/PV, exhaust fans and plenums, duct work, HEPA filters and flow switch)</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>NEC ITS</th>
<th>B</th>
<th>A</th>
</tr>
</thead>
<tbody>
<tr>
<td>TSR</td>
<td>Recept transportation window</td>
<td>Limitation on the radioactive inventory on the HVAC/PV/HEPA filters</td>
</tr>
<tr>
<td></td>
<td>Defence in depth</td>
<td>No leak path below the water line in the MCO</td>
</tr>
<tr>
<td></td>
<td>Cask vent line enters HVAC/PV piping away from operator</td>
<td></td>
</tr>
</tbody>
</table>

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<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator*</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4.0)</th>
<th>NRC ITS²</th>
</tr>
</thead>
<tbody>
<tr>
<td>E 2 Hydrogen explosion outside an MCO due to instrumentation failure</td>
<td>PB B 02a PB L 11f</td>
<td>Prevent hydrogen explosion Protect against instrumentation inaccuracy</td>
<td>Safety significant equipment for confinement and dilution HVAC/PV system (exhaust fans and plenums duct work HEPA filters and flow switch) HVAC/PV process hood isolation damper HVACD system (exhaust HEPA filter exhaust duct work, isolation damper) Standby electrical power (diesel generator and HVAC/PV system restart circuit) HVACB isolation dampers (outside inlets) Safety significant equipment for monitoring Reference air system (reference air header differential pressure alarms for process bays) TSR Limitation on the radionuclide inventory on the HVAC/PV HEPA filters Defense in depth equipment for detection SCIC process bay high temperature trip Defense in depth equipment for prevention Air dryer on the instrument air supply Defense in depth equipment for confinement, purge and pressurize Cask-MCO SCHe system Lines and valves to isolate and purge the MCO²</td>
<td>B</td>
</tr>
</tbody>
</table>
### Table B3-14: Summary of Safety Features Required to Mitigate or Prevent a Multi-Canister Overpack External Hydrogen Explosion (7 sheets)

<table>
<thead>
<tr>
<th>NRC TIS</th>
<th>B</th>
<th>B</th>
</tr>
</thead>
<tbody>
<tr>
<td>Safety features (described in Chapter B4(a))</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Safety significant equipment for confinement and dilution, HVAC/C/PV system (exhaust fans and plenums; duct work; HEPA filters and flow switch)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>HVAC/C/PV process hood isolation damper</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Standby electrical power (diesel generator and HVAC/C/PV system; restart circuit)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>HVAC/C/PV isolation damper (outside air mix)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>HVAC/C/PV system (exhaust HEPA filter; exhaust duct work; isolation damper)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Safety significant equipment for monitoring</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Reference air system (reference air header; differential pressure alarms for process bays)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>TSP Procedure to verify the results of the pressure rebound tests before continuing process steps</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Limitation on the radionuclide inventory on the HVAC/C/PV HEPA filters</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Defense in depth equipment for MCC process testing</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Initial pressure rebound test prior to proof of-dryness demonstration (temperature indicator on the tempered water [annular] system and pressure on the VPS)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Final pressure rebound test following the proof of-dryness demonstration after all potential water sources are isolated (temperature indicator on the tempered water [annular] system and pressure on the VPS)</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Checklist designation: PB 03b

Candidate accident: E3 Hydrogen explosion outside an MCO due to excessive water in MCO

The MCS will not allow out of sequence operation.

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B3-80
Table B3-14  Summary of Safety Features Required to Mitigate or Prevent a Multi-Canister Overpack External Hydrogen Explosion (7 sheets)

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator*</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4 0)</th>
<th>NRC ITSb</th>
</tr>
</thead>
<tbody>
<tr>
<td>E 4 Hydrogen explosion outside an MCO due to process upset of key parameters</td>
<td>PB B 03a</td>
<td>Prevent hydrogen accumulation and explosion</td>
<td>Safety significant equipment for confinement and dilution</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td>PB B 13a</td>
<td></td>
<td>HVAC/PV system (exhaust fans and plenums duct work, HEPA filters and flow switch)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>PB B 13b</td>
<td></td>
<td>HVAC/PV process hood isolation damper</td>
<td></td>
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<tr>
<td></td>
<td>PB B 13c</td>
<td></td>
<td>HVACD system (exhaust HEPA filter exhaust duct work, isolation damper)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>PB H 06f</td>
<td>Protect against instrument inaccuracy</td>
<td>Standby electrical power (diesel generator and HVAC/PV system restart circuit)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>PB H 06g</td>
<td></td>
<td>HVACB isolation dampers (outside air inlets)</td>
<td></td>
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<tr>
<td></td>
<td>PB H 11d</td>
<td></td>
<td>Safety significant equipment for monitoring</td>
<td></td>
</tr>
<tr>
<td></td>
<td>PB H 11e</td>
<td></td>
<td>Reference air system (reference air header, differential pressure alarms for process bays)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>PB L 11d</td>
<td></td>
<td>TSR</td>
<td>B</td>
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<tr>
<td></td>
<td>PB L 11e</td>
<td></td>
<td>Limitation on the radionuclide inventory on the HVAC/PV HEPA filters</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>Defense in depth equipment for detection of process upset</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>Tempered water (annulus) system temperature trip</td>
<td></td>
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<td></td>
<td></td>
<td></td>
<td>General service helium system safety-class flow instrumentation</td>
<td></td>
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<td></td>
<td></td>
<td></td>
<td>SCIC vacuum cycle timer (8 4 4)</td>
<td></td>
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<td></td>
<td>VPS instrumentation (pressure)</td>
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<td></td>
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<td></td>
<td>SCIC process bay high temperature trip</td>
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<td></td>
<td></td>
<td></td>
<td>Tempered water (annulus) system, tempered water temperature trip and tempered water level alarm</td>
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<td></td>
<td>Defense in depth equipment for prevention</td>
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<td></td>
<td>Air dryer on instrument air supply</td>
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<td></td>
<td>Defense in depth equipment for confinement and shutdown</td>
<td></td>
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<tr>
<td></td>
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<td></td>
<td>Cask-MCO</td>
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<td></td>
<td></td>
<td></td>
<td>SCHe system</td>
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<td></td>
<td>Lines and valves to isolate and purge the MCO</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>Tempered water (annulus) system</td>
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</tr>
</tbody>
</table>

November 1999
Table B3-14  Summary of Safety Features Required to Mitigate or Prevent a Multi-Canister Overpack External Hydrogen Explosion  (7 sheets)

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4 0)</th>
</tr>
</thead>
<tbody>
<tr>
<td>E 5 Hydrogen explosion outside an MCO due to loss of support utilities</td>
<td>PB F 02a, PB F-05, SB F 01b, SB F 02b, OU P 04, OU R 02, OU R 03, OU R 04</td>
<td>Prevent hydrogen explosion</td>
<td>Safety significant equipment for confinement and dilution</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>VPS components (pressure management)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>HVAC/PV system (exhaust fans and plenums duct work, HEPA filters and flow switch)</td>
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<td></td>
<td></td>
<td></td>
<td>HVAC/PV process hood isolation damper</td>
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<td></td>
<td></td>
<td></td>
<td>Standby electrical power (diesel generator and HVAC/PV system restart circuit)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>HVACB isolation dampers (outside air inlets)</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>HVACD system (exhaust HEPA filter exhaust duct work, isolation damper)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Place the MCO in a safe configuration during a loss of support utilities</td>
<td>Safety significant equipment for monitoring</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Reference air system (reference air header differential pressure alarms for process bays)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>TSR</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Recept transportation window</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Limitation on the radionuclide inventory on the HVAC/PV HEPA filters</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Administrative procedure for restricting crane movement during MCO processing except as part of an approved recovery plan</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Safety class equipment (providing a safety significant function) for confinement and dilution</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Cask-MCO</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>SCHe system (including vent delay)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Lines and valves to isolate and purge the MCO</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Defense in depth equipment for detection of process upset</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Safety-class detection of process upset and safety-class equipment (performing a safety significant function) for shutdown designed to fail to a safe position</td>
</tr>
</tbody>
</table>

NRCITSb: B
<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4 0)</th>
<th>NRC ITSa</th>
</tr>
</thead>
<tbody>
<tr>
<td>E 6 Hydrogen explosion outside an MCO due to facility fire</td>
<td>TC J 12, PB L 01, PB L 02, PB L 03, PB L 04, PB L 05, PB L 06, PB L 07, PB L 08, PB L 09, PB L 10, PB L 13, PB L 14, PB L 15, PB L 16, PB P-02, OU P 02a</td>
<td>Protect a processing area from a fire that could spread to the process bay</td>
<td>Safety class equipment (performing a safety significant function) for detection&lt;br&gt;SCIC process bay high temperature trip&lt;br&gt;TSR&lt;br&gt;Combustible loadings limited&lt;br&gt;Limitation on the radionuclide inventory on the HVACC/PV HEPA filters&lt;br&gt;Restore bay temperatures following process bay high temperature trip&lt;br&gt;Defense in depth equipment for detection&lt;br&gt;SCIC process bay temperature detection&lt;br&gt;Defense in depth equipment for prevention&lt;br&gt;Air dryer on the instrument air supply&lt;br&gt;Defense in depth equipment for confinement, purge and pressurize&lt;br&gt;Cask-MCO&lt;br&gt;SCH system&lt;br&gt;Lines and valves to isolate and purge the MCO&lt;br&gt;Defense in depth fire equipment&lt;br&gt;Programmatic controls for fire prevention&lt;br&gt;Fire protection system present in each bay&lt;br&gt;TSR&lt;br&gt;Shipment paperwork verified for helium supply gas content during receipt</td>
<td>B</td>
</tr>
<tr>
<td>E 7 Hydrogen explosion outside an MCO due to contamination of helium supply</td>
<td>PB H 06k</td>
<td>Prevent contamination of helium supply</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Table B3-14  Summary of Safety Features Required to Mitigate or Prevent a
Multi-Canister Overpack External Hydrogen Explosion  (7 sheets)

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designtor*</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4 0)</th>
<th>NRC ITSb</th>
</tr>
</thead>
<tbody>
<tr>
<td>E 8 Hydrogen explosion outside an MCO due to line break caused by a seismic event</td>
<td>TC R 01 PB R 01a OU R 01a</td>
<td>Control materal at risk</td>
<td>No SSCs required TSR Limitation on the radionuclide inventory on the HVAC/PV HEPA filters</td>
<td></td>
</tr>
</tbody>
</table>

*Checklist designators are from HIE SD SNF HIE 004 1999 Cold Vacuum Drying Facility Hazard Analysis Rev 4 Fluor Daniel Hanford Incorporated, Richland, Washington

b US Nuclear Regulatory Commission important to-safety classifications Category A = critical to safe operation, Category B = major impact on safety Category C = minor impact to safety

*Lines and valves to isolate the MCO include the isolation valves (and filters on air supply to valve actuators) in the VPS general service helium system, PWC system, and SCHe system.

HEPA = high-efficiency particulate air (filter)
HVAC = heating, ventilation, and air conditioning
HVACB = process bay recirculation HVAC system
HVACC/PV = process bay local exhaust HVAC and process vent system.
HVACD = process general supply/exhaust HVAC system
ITS = important to safety
MCO = multi-canister overpack.
MCS = monitoring and control system.
NRC = US Nuclear Regulatory Commission
PWC = process water conditioning
SCHe = safety-class helium
SCIC = safety class instrumentation and control
SSC = structure system and component
TSR = technical safety requirement.
VPS = vacuum purge system

8 4 4 = 8 hour initial vacuum cycle 4 hour subsequent vacuum cycles 4 hour return to pressure between vacuum cycles
E 2 Hydrogen explosion outside an MCO due to instrumentation failure

The accident scenarios identified in the hazard analysis for the E2 bin represent external hydrogen explosions caused by elevated bay temperatures (and subsequent potential for elevated fuel temperatures and reaction rates) and actuation of the SCHe system concurrent with a no-flow condition in the local exhaust (HNF-SD-SNF-HIE-004). The controls in the E2 bin prevent the external hydrogen explosion by providing dilution in the local exhaust system and mitigation by limiting the material at risk on the local exhaust HEPA filters. In addition, the general exhaust system and SSCs that support process bay confinement are credited because the local exhaust cannot be guaranteed to provide confinement in all cases following the external explosion.

- Process bay local exhaust HVAC and process vent system process hood isolation damper and instrument air supply

  Isolation dampers in the process bay local exhaust HVAC and process vent system process hood fail closed. If power is lost, dampers will open with electrical power from the standby power system and instrument air supplied by the local dedicated tank. The hood isolation damper and instrument air supply operate in conjunction with the standby power system to facilitate HVAC operating while on standby power.

- Process general supply/exhaust HVAC system (exhaust HEPA filter, exhaust duct work, isolation damper)

  The process general supply/exhaust HVAC system mitigates a release into the process bay by sweeping it through HEPA filters before discharging it outside the facility. The process general supply/exhaust HVAC system also provides confinement in conjunction with the facility's structure by maintaining a negative building pressure. Fail-closed exhaust dampers from the process bays isolate other flow paths and ensure that differential pressure is maintained.

- Reference air system (reference air header, differential pressure alarms)

  The reference air system monitors the negative pressure in the process bays and process water tank room by providing differential pressure indication and alarms to the control room for operator response.

- Standby electrical power (diesel generator and process bay local exhaust HVAC and process vent system restart circuit)

  The standby power system provides connections to restart the local exhaust fans and supporting equipment. Operation of the local exhaust on standby power will maintain building differential pressure sufficient for confinement during facility power outages.
Process bay recirculation HVAC system isolation dampers (outside inlets)

The process bay recirculation HVAC system provides fail-closed outside air inlet dampers so the local exhaust on standby power can maintain process bay differential pressure.

E 3 Hydrogen explosion outside an MCO due to excessive water in MCO

The accident scenarios identified in the hazard analysis for the E 3 bin represent external hydrogen explosions caused by excessive water remaining in an MCO (instrumentation failure or operator error) (HNF-SD-SNF-HIE-004). The controls in the E3 bin prevent the external hydrogen explosion by providing dilution in the local exhaust system and by preventing excessive free water in an MCO at an unexpected time. Mitigation is provided by a limitation on the material at risk on the local exhaust HEPA filters. In addition, the general exhaust system and SSCs that support process bay confinement are credited because the local exhaust cannot be guaranteed to provide confinement in all cases following the external explosion.

- All HVAC system equipment identified for accident E 2

Assumptions made that require protection by TSRs are listed below

- Procedure to verify the results of the pressure rebound tests before continuing process steps

An initial pressure rebound test surveillance (pressure rise test) must be met before entry into the proof-of-dryness demonstration is allowed. Similarly, a proof-of-dryness demonstration surveillance must be met before the final pressure rebound test steps can begin. Finally, a final pressure rebound test must be met before shipment preparation steps can begin.

E 4 Hydrogen explosion outside an MCO due to process upset of key parameters

The accident scenarios identified in the hazard analysis for the E 4 bin represent external hydrogen explosions caused by elevated hydrogen generation rates (high temperature tempered water, water ingress into MCO, air ingress into MCO, loss of annulus water), venting of MCO before breakthrough during draining, and venting cask-MCO while preparing for shipping (HNF-SD-SNF-HIE-004). The controls in the E4 bin prevent the external hydrogen explosion by providing dilution in the local exhaust system and mitigation by limiting the material at risk on the local exhaust HEPA filters. In addition, the general exhaust system and SSCs that support process bay confinement are credited because the local exhaust cannot be guaranteed to provide confinement in all cases following the external explosion.

- All HVAC system equipment identified for accident E 2
E 5 Hydrogen explosion outside an MCO due to loss of support utilities

The accident scenarios identified in the hazard analysis for the E 5 bin represent external hydrogen explosions caused by a loss of a support system (e.g., instrument air) because of accidents in adjacent bays, crane load drops, and accidents in the spare bay. This bin also includes loss of power events caused by external forces (e.g., vehicle accident), flooding, and lightning strike (HNF-SD-SNF-HIE-004). The controls in the E 5 bin prevent an external hydrogen explosion by providing dilution in the local exhaust system and by controlling specific initiators that result in a loss of support utilities (e.g., crane load drop and facility loss of power). In addition, the general exhaust system and SSCs that support process bay confinement are credited because the local exhaust cannot be guaranteed to provide confinement in all cases following the external explosion.

- All HVAC system equipment identified for accident E 2
- SCHe system (vent delay)
  The pressure-regulated discharge flow path from the MCO to the local exhaust system ensures a minimum 1 0-minute delay from SCHe initiation until discharge flow initiation to the local exhaust to allow standby power restart of the local exhaust
- Lines and valves to isolate and purge the MCO
  Lines and valves for isolating the MCO include the isolation valves (and filters on air supply to valve actuators) in the VPS, general-service helium system, PWC system, and SCHe system. Upon demand, all the valves close to isolate the MCO, except the SCHe system valves, which open to allow helium to the MCO. These SSCs ensure the proper functioning of the SCHe system.
- VPS components (pressure management)
  The 30 lb/in² gauge check valve on the 30 lb/in² gauge pressure relief line (backup pressure management to the SCHe system) opens, vents a small amount of hydrogen, and then re-seats to limit hydrogen flows into the local exhaust duct. The valve prevents a blowdown of the MCO from 30 lb/in² gauge to atmospheric pressure and the corresponding release of a large volume of hydrogen.

Assumptions made that require protection by TSRs are listed below:

- Administrative procedure for restricting crane movement during MCO processing except as part of an approved recovery plan
  Bridge crane movement is not allowed from the time that draining operations begin until the proof-of-dryness demonstration is complete and the MCO isolation valves...
are closed unless as part of an approved recovery plan. This restriction ensures that key process lines cannot be sheared by crane movement, which could cause failure of features that provide hydrogen control.

E 6 Hydrogen explosion outside an MCO due to facility fire

The accident scenarios identified in the hazard analysis for the E 6 bin represent external hydrogen explosions caused by an internal or external facility fire, including a fire in the transfer corridor caused by maintenance activities (HNF-SD-SNF-HIE-004). The controls in the E 6 bin prevent an external hydrogen explosion by ensuring that control of the MCO and process is not lost in the event of a fire in the process bay. The control on combustibles precludes damage to safety-related SSCs whose malfunction could result in the loss of control of the MCO or process. The control on combustibles necessitates protection of the initial bay temperature, which is an assumption made in the fire hazards analysis.

Safety-class equipment (performing a safety-significant function) to prevent an external hydrogen explosion:

- SCIC process bay high temperature trip

  The SCIC process bay high temperature trip acts to protect an initial condition assumption in the fire hazards analysis.

Assumptions made that require protection by TSRs are listed below:

- Combustible loadings limited

  While an MCO is present in the facility, combustible loadings are limited as determined by the fire hazard analysis implementation plan (SNF-4942). These limits ensure that any fire in the CVDF does not result in uncontrolled releases (e.g., fire-caused loss of process control).

- Restore bay temperature following a process bay high temperature trip

  On high bay temperature trip alarm, operations must return the process bay temperature to within acceptable limits.

E 7 Hydrogen explosion outside an MCO due to contamination of helium supply

The accident scenarios identified in the hazard analysis for the E 7 bin represent external hydrogen explosions caused by contaminants in the helium supply (unexpected fuel reactions) (HNF-SD-SNF-HIE-004). The controls in the E 7 bin prevent an external hydrogen explosion by controlling the helium supply.
Assumptions made that require protection by TSRs are listed below:

- Shipment paperwork for gas bottle content during receipt

  The manufacturer's paperwork and shipping papers will be checked upon helium cylinder receipt at the CVDF, both the normal and safety-class supply, to verify that the cylinder's contents were sampled by the supplier and that the sample met the required purity specification of >99% helium.

E 8  Hydrogen explosion outside the MCO due to line break caused by a seismic event

  The accident scenarios identified in the hazard analysis for the E 8 bin represent external hydrogen explosions in the transfer corridor or process bay caused by a seismic event (HNF-SD-SNF-HIE-004). A seismic event is an unlikely initiating event, therefore no credit is taken for preventative features in the E8 bin. Credit is taken for limiting the material at risk on the local exhaust HEPA filters.

No additional requirements result from analysis of this accident.

B3 4 2 4  Multi-Canister Overpack Internal Hydrogen Explosion  This section addresses the consequences of an MCO internal hydrogen explosion. The quantity of radioactive material released is calculated and that value is used to calculate the dose consequences to both offsite and onsite receptors.

  The unmitigated dose consequences of the bounding MCO internal hydrogen explosion do not exceed the offsite release limit, so no safety-class features are required to mitigate or prevent this accident. However, the dose consequences exceed the onsite risk evaluation guideline, so safety-significant features have been identified to mitigate the consequences of an internal hydrogen explosion. Components of the SCIC, VPS, general-service helium, tempered water (annulus), and SCHe systems function to prevent MCO internal hydrogen explosion accidents. For this accident category, these safety-class systems are functioning in a safety-significant prevention role. The safety-significant portions of the process bay local exhaust HVAC system (including standby power), the process general exhaust HVAC system, and the differential pressure alarms provide an additional barrier to mitigate a release into the process bays during the postulated accident scenario. Details of the calculations for this DBA are provided in SNF-2770, Chapter 5.

B3 4 2 4 1  Scenario Development  Because hydrogen is expected to be generated in the MCO during processing at the CVDF, the cold vacuum drying process incorporates many features to remove hydrogen and prevent the ingress of oxygen (air) into the MCO. The primary method of maintaining an oxygen-free atmosphere and a low hydrogen concentration in the MCO is to inert the MCO with helium before shipping and to pressurize and purge the MCO with helium at appropriate stages of the CVDF process. Process monitoring during drying is used to ensure that hydrogen is not building up in the system. Therefore, conditions that could result in significant quantities of oxygen and hydrogen accumulating inside the MCO during the draining,
drying, and proof-testing portions of the process require multiple failures of equipment and process monitoring.

With no helium fed into and no helium or hydrogen bled from the system to prevent hydrogen buildup, hydrogen gas concentration in the MCO and piping system increases with time. The bounding accident scenario postulates a line break that occurs when the MCO is under partial vacuum. Either process line, the one connected to the long axial process tube or the one connected to the filtered process exit port, could fail and provide the source of the air ingress. Since a motive force is needed to draw the air into the MCO, the event is postulated to occur under worst-case vacuum conditions (e.g., early in the drying process).

**B3 4 2.4.2 Source Term Analysis** The following assumptions are used in the analysis:

- A vacuum pressure of 0.3 atm (228 torr) exists in the MCO when the line break occurs. This pressure allows enough air to be drawn into the MCO to form a relatively large quantity of gas having a two-to-one mixture of hydrogen to oxygen (a stoichiometric ratio). This mixture yields a maximum energy release when it is ignited.

- An ignition source in the MCO exists. This source could be from rapid oxidation of uranium hydride or a static electrical discharge.

- Complete combustion takes place, that is, all the hydrogen reacts with all the oxygen to produce water.

- The average fuel temperature is assumed to be 75 °C (167 °F) so that the hydrogen generation rate is adequate to produce nearly pure hydrogen gas in the MCO at a pressure of 0.3 atm within 8 hours (SNF-2770).

- The MAR includes 15 kg of particulate generated between the time fuel washing is completed at the K Basins and draining operations are completed at the CVDF and 10 kg of particulate generated from the start of drying up to the time of the explosion (see Section B3 4 2). The total MAR, 25 kg of particulate treated as UO₂, represents approximately 22 kg of uranium (unit doses are calculated on a per-gram-of-uranium basis).

- It is assumed that particulate not removed by washing at K Basins (21 kg) will not be released by the hydrogen explosion. The conservatism associated with the MAR calculation and ARF selection is adequate to bound the potential for additional particulate removal by detonation shockwaves in localized areas in the MCO. The particulate is not uniformly distributed over the surface of the fuel but is located in cracks and crevices that would not be exposed to the detonation shockwaves.

Reaction of water with uranium metal is the main source of hydrogen. Other sources are from radiolysis and the degradation of uranium hydride. The hydrogen generation rate for fuel
and water at a temperature of 75 °C (167 °F) is calculated to be several moles per hour (SNF-2770). This hydrogen generation rate is adequate to flush most of the helium from the MCO. It also is assumed that a vacuum pressure of 0.3 atm (228 torr) has been achieved in the MCO when the line break occurs. This pressure produces a stoichiometric mixture of hydrogen and oxygen after the air enters the MCO. These conditions give maximum values for combustion energy release.

In the unmitigated scenario, when the mixture ignites, the resulting deflagration or detonation pressurizes the MCO and particulate matter is blown out through the line break or a rupture disk. The first concern is to determine whether the pressure impulse from the hydrogen combustion event can catastrophically fail the MCO vessel boundary. For the purposes of this analysis, the event is conservatively treated as a detonation with a shock wave at twice the adiabatic combustion pressure. This will occur for hydrogen concentrations of approximately 30% by volume. The worst-case peak pressure in the MCO following a hydrogen burn could exceed the 150 lb/in² gauge or 30 lb/in² gauge rupture disk pressure (SNF-2770) but will not catastrophically fail the MCO. The MCO can withstand static pressures in excess of 340 lb/in² gauge (HNF-SD-SNF-SARR-005) with the limiting components being the threads on the shield plug and MCO collar. The MCO shell can withstand a pressure of 1,790 lb/in² gauge. The peak reflected pressure from a detonation shockwave could reach 530 lb/in² gauge (SNF-2770), but such a pressure pulse would not exist long enough to allow for complete relaxation of the MCO collar or shield plug threads, therefore, catastrophic failure would not occur. The peak equilibrium pressure due to thermodynamic effects following the burn is approximately 110 lb/in² gauge, which does not challenge the MCO boundary. Because of the possible multiple release pathways (rupture disks and the line leak that allowed the air ingress), no holdup of the source term in the MCO is assumed (i.e., the entire source term is released).

The MCO is pressurized by heating the gas. The MCO will vent for as long as it takes for the pressure in the MCO to return to atmospheric pressure. With complete combustion of hydrogen, the gas will consist of steam, nitrogen, helium, and radioactive particulate matter that is aerodynamically entrained by the explosion. DOE-HDBK-3010-94, Section 4 4 2 3 2 gives an overall respirable release factor of $2 \times 10^3$ for venting of pressurized gas through UO₂ powders (see Section B3 4 2 6). DOE-HDBK-3010-94 presents two choices for release fractions under these conditions, one for low pressure (less than 25 lb/in² gauge) and one for relatively high pressures (greater than 25 lb/in² gauge). The release fraction for low pressure conditions was selected even though failure occurs at higher pressures. This is because the experiments investigating the release fraction were performed under high air flow conditions, whereas the powder in the MCO will be exposed to very low air velocities. Thus the smaller release fraction, $2 \times 10^3$, was assumed in this analysis. This value bounds the effects of shock, blast, and venting. The source term is the product of the expelled volume fraction, the MAR, and the release factor.

$$
(22 \text{ kg}) \left( \frac{1,000 \text{ g}}{1 \text{ kg}} \right) (2 \times 10^3) = 44 \text{ g U}
$$
B3 4 2 4 3 Consequence Analysis  The radiological dose (effective dose equivalent) to a receptor is calculated by using the following equation

\[ DE = M \times \frac{\chi}{Q'} \times BR \times UD \]

where

- \( DE \) = effective dose equivalent based on inhalation exposure only (rem)
- \( M \) = respirable particulate quantity released into the air (grams)
- \( \chi/Q' \) = air transport factor (s/m³)
- \( BR \) = average inhalation rate during the release (m³/s)
- \( UD \) = committed effective dose equivalent per unit gram inhaled

The respirable particulate quantity is the 44 g of fuel calculated in Section B3 4 2 4 2. The value for \( \chi/Q' \) is that shown in Table B3-15. The low activity breathing rate is \( 3.33 \times 10^{-4} \) m³/s as defined in HNF-SD-SNF-TI-059. The dose per unit of respirable material inhaled is \( 4.38 \times 10^5 \) rem/g of fuel as specified in Section 3 4 1 2 of the SNF Project FSAR.

Unmitigated Consequences  The unmitigated dose to the onsite receptor is calculated as follows

\[ D_{on-site} = M \times \frac{\chi}{Q'} \times BR \times UD \]

\[ = (44 \text{ g})(7.32 \times 10^{-2} \text{ s/m}^3)(3.33 \times 10^{-4} \text{ m}^3/\text{s})(4.38 \times 10^5 \text{ rem/g}) \]

\[ = 470 \text{ rem (4.7 Sv)} \]

The dose consequences at the remaining receptor sites are calculated in the same manner and are shown in Table B3-15.

Mitigated Consequences  Mitigation of the internal hydrogen explosion accident takes credit for the integrity of selected isolation valves and lines to isolate the MCO, the SCIC system and related instrumentation in other systems, the \( \text{SCH}_e \) system, the tempered water (annulus) system, and the cask–MCO. The isolation valves and lines inside the safety-class portion of the VPS and PWC systems ensure that there is no air ingress into the MCO. The SCIC system provides control functions to initiate isolation and purge of the MCO (isolation and purge trip) if the process upsets exceed trip setpoints. The \( \text{SCH}_e \) system provides a positive means of providing a minimum dedicated helium purge to the MCO to sweep hydrogen or air (if an air ingress were to occur) from the MCO, in the event that general-service helium were not available. The tempered water (annulus) system ensures that MCO temperatures, and subsequent hydrogen generation rates, remain within analyzed bounds.
Table B3-15 Dose Calculation Summary for Hydrogen Combustion within a Multi-Canister Overpack

<table>
<thead>
<tr>
<th>Receptor location (distance direction)</th>
<th>Duration (hours)</th>
<th>$\chi/Q^a$ (s/m$^3$) (without plume meander)</th>
<th>Unmitigated dose$^b$ (rem (Sv))</th>
<th>Unmitigated evaluation guideline/release limit$^c$ (rem (Sv))</th>
<th>Mitigated dose (Sv)</th>
<th>Mitigated evaluation guideline/release limit$^d$ (Sv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Onsite (100 m E)</td>
<td>&lt;1</td>
<td>732 E-02</td>
<td>470 (4.7)</td>
<td>10 (1.0 E-02)</td>
<td>Prevented$^d$</td>
<td>Not applicable accident prevented</td>
</tr>
<tr>
<td>Columbia River near bank (650 m W)</td>
<td>&lt;1</td>
<td>244 E-03</td>
<td>16 (0.16)</td>
<td></td>
<td>Prevented$^d$</td>
<td>--</td>
</tr>
<tr>
<td>100 Area Fire Station (3 750 m ESE)</td>
<td>&lt;1</td>
<td>160 E-04</td>
<td>10 (0.01)</td>
<td></td>
<td>Prevented$^d$</td>
<td></td>
</tr>
<tr>
<td>Hanford Site boundary (10 090 m W)</td>
<td>&lt;1</td>
<td>448 E-05</td>
<td>0.29 (0.0029)</td>
<td>0.5 (5.0 E-03)</td>
<td>Prevented</td>
<td>Not applicable accident prevented</td>
</tr>
</tbody>
</table>


$^a$Fifty year committed effective dose equivalent.

$^b$Evaluation guideline (for onsite receptor) and release limit (for offsite receptor) for unmitigated accidents in the anticipated frequency category (i.e., frequency $>10^{-2}$ per year)

$^c$The features designed to prevent this accident result in an estimated accident frequency in the beyond extremely unlikely range (i.e., $<10^{-4}$ per year)
Since the unmitigated accident does not exceed the offsite dose limits, no mitigated consequences were calculated for the offsite receptor. The unmitigated accident leads to onsite doses that are greater than the risk evaluation guideline for anticipated events. With the credited controls, the accident is prevented to beyond extremely unlikely. In addition, the presence of HEPA filters reduces the onsite dose for the confinement boundary failure postulated in the scenario.

**B3 4.2.4.4 Comparison to Guidelines**

**Comparison of Unmitigated Doses** Event tree sequences indicate that the unmitigated frequency for the internal hydrogen explosion DBA is in the anticipated category (i.e., an unmitigated frequency greater than $10^{-2}$ per year) (see SNF-2770, Appendix A). The unmitigated sequence considered the processing of 200 MCOs per year. The unmitigated radiological offsite dose for this event is below offsite release limits, while the unmitigated onsite dose for an anticipated event is above onsite risk evaluation guidelines.

**Comparison of Mitigated Doses** The mitigated frequency of the event tree sequence that represents this DBA as an internal hydrogen explosion is $3 \times 10^{-7}$ per year, thus this mitigated DBA is beyond extremely unlikely (see SNF-2770, Appendix A). With safety-class and safety-significant features credited (performing a safety-significant function), the mitigated internal hydrogen explosion is both prevented and mitigated.

**B3 4.2.4.5 Summary of Safety-Class Structures, Systems, and Components and Technical Safety Requirement Controls** Under normal operating conditions, helium flow through the MCO precludes the accumulation of flammable concentrations of hydrogen in the MCO. Under upset or accident conditions, safety-significant equipment is required in order to ensure this capability is not lost. The function of precluding flammable concentrations could be compromised by a loss of helium flow or an air ingress.

The checklist designators included in the accident bins, other than the accident selected as the DBA, represent additional accident sequences slightly different than the DBA. All of these binned accidents are bounded by the DBA because they have lesser or equivalent worst-case consequences and frequencies.

The accident scenarios identified in the hazard analysis for the I 1 bin represent internal hydrogen explosions caused by elevated fuel corrosion rates (high temperature tempered water), water ingress into MCO, and air ingress into MCO (HNF-SD-SNF-HIE-004). The controls in the I 1 bin prevent an internal hydrogen explosion by controlling the hydrogen concentration in the MCO and by preventing air ingress into the MCO. The normal hydrogen concentrations inside the MCO are controlled to ensure that if the MCO gas contents were to be mixed with air, the resulting gas mixture would not be flammable. The likelihood of an internal explosion is further reduced by crediting features that prevent air ingress so that onsite risk evaluation guidelines and offsite release limits are satisfied. The SSCs that limit fuel temperature also are credited to protect the analysis assumption pertaining to hydrogen generation rates. In addition, because the scenario postulates a process line break that is tantamount to a gaseous release, confinement.
systems are credited with controlling that release. Note that the confinement systems are not credited with mitigation of the internal explosion release but only with mitigation of the gaseous release associated with the line break.

The safety-significant equipment and controls designated to prevent the dose consequences of the MCO internal hydrogen combustion accident are described below.

Safety-class equipment (performing a safety-significant function) for detection of process upset:

- **SCIC system including vacuum cycle timer (8-4-4 logic)**
  
The SCIC system uses programmable logic controllers, wiring to process instrumentation, vacuum limit timer, signals from seismic detectors and temperature monitors, system controls, and output relays to perform an isolation and purge trip and a tempered water trip.

- **General-service helium system safety-class flow instrumentation**
  
The general-service helium system provides helium flow information to the SCIC system to initiate MCO isolation and SCHe actuation during process upset conditions.

- **Vacuum purge system pressure instrumentation**
  
Pressure indicators on the vacuum purge system provide MCO internal pressure information to the SCIC system, which initiates MCO isolation and SCHe actuation during process upset conditions.

- **Tempered water (annulus) system temperature trip**
  
The tempered water (annulus) system includes antisiphon valves, low water level indication and alarm, and manual refill capability to ensure a minimum water level is maintained above the elevation of the fuel within the MCO. The system detects high water temperature and signals the SCIC system to actuate the tempered water trip.

- **Tempered water (annulus) system low level alarm**
  
Redundant liquid level indicators are used by the tempered water (annulus) system to detect a low water level in the cask–MCO annulus and to provide a signal to the SCIC system to actuate the low-level alarm.

- **Tempered water (annulus) system level check petcocks**
The level check petcocks provide a manual means of verifying the presence of water in the annulus. The petcocks are located in and above the double-walled portion of the tempered water (annulus) piping and are connected to the outer and inner pipes above the height of the fuel in the MCO. If a low-level alarm is received, the petcocks can be used to verify the presence of water in the inner or outer pipe, thus indicating the presence of water in the annulus.

Safety-class equipment (performing a safety-significant function) for confinement, purge, and pressurization

- **Cask-MCO**

The cask-MCO is a major part of the pressure boundary for confinement of radioactive materials during processing and provides connections for the process piping to the SCHe system. The MCO also acts as the vessel for helium dilution of hydrogen.

- **SCHe system**

The SCHe system provides two redundant and independent paths for purging and pressurizing the MCO and venting to the process vent. The SCHe purging and pressurizing functions prevent flammable concentrations of hydrogen and oxygen from forming within the MCO.

- **Tempered water (annulus) system piping and antisiphon valves**

The tempered water system provides for sufficient heat transfer from the MCO so that excessive corrosion rates and related hydrogen generation rates are not experienced.

- **Lines and valves to isolate and purge the MCO**

Lines and valves to isolate the MCO include the isolation valves (and filters on air supply to valve actuators) in the VPS general-service helium system, PWC system, and SCHe system. Upon demand, all the valves close to isolate the MCO, except the SCHe system valves, which open to allow helium to the MCO. These SSCs ensure the proper functioning of the SCHe system.

Safety-significant equipment for confinement

- **Process bay local exhaust HVAC and process vent system (exhaust fans and plenums, duct work, HEPA filters)**

The process bay local exhaust HVAC and process vent system mitigates a gaseous release (that would occur subsequent to the line break postulated in this event) into
the process bay by sweeping it through HEPA filters before it is discharged outside the facility

- Process bay local exhaust HVAC and process vent system process hood isolation damper and instrument air supply

Isolation dampers in the process bay local exhaust HVAC and process vent system process hood fail closed. If power is lost, dampers will open with electrical power from the standby power system and instrument air supplied by the local dedicated tank. The hood isolation damper and instrument air supply operate in conjunction with the standby power system to facilitate HVAC operating while on standby power.

- Process general supply/exhaust HVAC system (exhaust HEPA filter, exhaust duct work, isolation damper)

The process general supply/exhaust HVAC system mitigates a release into the process bay or process water tank room by filtering it before discharging it outside the facility. The process general supply/exhaust HVAC system also provides confinement in conjunction with the facility’s structure by maintaining a negative building pressure. Fail-closed exhaust dampers from the process bays and process water tank room isolate other flow paths and ensure that differential pressure is maintained.

- Reference air system (reference air header, differential pressure alarms)

The reference air system monitors the negative pressure in the process bays and process water tank room by providing differential pressure indication and alarms to the control room for operator response.

- Standby electrical power (diesel generator and process bay local exhaust HVAC and process vent system restart circuit)

The standby power system provides connections to restart the local exhaust fans and supporting equipment. Operation of the local exhaust on standby power will maintain building differential pressure sufficient for confinement during facility power outages.

- Process bay recirculation HVAC system isolation dampers (outside air inlets)

The process bay recirculation HVAC system provides fail-closed outside air inlet dampers so the local exhaust on standby power can maintain process bay differential pressure.

Assumptions made that require protection by TSRs are listed below.

- Perform process connector leak test before processing
The VPS process connectors must be verified to be leak tight to protect analysis assumptions in relation to air inleakage into the MCO and process systems. In the frequency analysis, credit is taken for a reduced likelihood of an air inleakage upon a leak test of the process connectors.

- Close long axial process tube port before closing filtered process exit port at the end of processing.

When isolating the MCO at the end of processing, the long axial process tube port must be closed at least 5 minutes before closing the filtered process exit port. This ensures that an SCIC trip will occur if the procedure to isolate the MCO is inadvertently conducted on an MCO still to be processed. Under some conditions, if the filtered process exit port is closed first on an MCO still being processed, it is possible to bypass SCIC trips.

The bounding MCO internal hydrogen explosion accident (I1) and the other accidents identified in the CVDF hazard analysis report (HNF-SD-SNF-HIE-004) that can potentially involve hydrogen combustion in an MCO are itemized in Table B3-16, along with corresponding checklist designators from the hazard analysis report, safety functions, and SSCs.

The other accidents within the remaining bins require the following safety SSCs and TSRs in addition to the ones identified for the DBA (I1):

- I2 Hydrogen explosion within an MCO due to instrumentation failure (significant air ingress into the MCO)

The accident scenario identified in the hazard analysis for the I2 bin represents an internal hydrogen explosion caused by elevated fuel corrosion rates (high bay temperature) (HNF-SD-SNF-HIE-004). The controls in the I2 bin prevent an internal hydrogen explosion by protecting against the malfunction of SSCs. Upon detection of elevated temperatures in the process bay, the MCO is placed into a safe and stable configuration to remove reliance on non-safety related SSCs. For an internal explosion to be possible, a line leak must have occurred to allow an air ingress. To mitigate the gaseous release associated with a line leak, credit is taken for confinement systems to control that release. Note that the confinement systems are not credited with mitigation of the internal explosion release but only with mitigation of the gaseous release associated with the line leak.

Safety-class equipment (performing a safety-significant function) for detection:

- SCIC process bay high temperature detection

The SCIC high bay temperature trip isolates the MCO and actuates the SCHe system so excessive temperatures in the bay cannot cause instrument inaccuracies or malfunctions that might result in MCO overpressurization.
Table B3-16  Summary of Safety Features Required to Mitigate or Prevent a Multi-Canister Overpack Internal Hydrogen Explosion  (6 sheets)

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator*</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4 0)</th>
<th>NRC ITS⁹</th>
</tr>
</thead>
<tbody>
<tr>
<td>11 Hydrogen explosion with an MCO due to process upset of key parameters (bounding accident, PB B 13a)</td>
<td>PB B-03a</td>
<td>Prevent hydrogen explosion</td>
<td>Safety class equipment (performing a safety significant function) for detection of process upset</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td>PB B 13a</td>
<td>Prevent accumulation of flammable concentrations of hydrogen Protection against air ingress</td>
<td>SCIC including vacuum limit timer General service helium system safety class flow instrumentation VPS pressure instrumentation Tempered water (annulus) system temperature trip Tempered water (annulus) low level Tempered water (annulus) level check petcocks</td>
<td></td>
</tr>
<tr>
<td></td>
<td>PB B 13b</td>
<td>Safety class equipment (performing a safety significant function) for confinement, purge and pressurize</td>
<td>Cask-MCO SCHe system Lines and valves to isolate and purge the MCO Tempered water (annulus) piping and antisphon valves</td>
<td></td>
</tr>
<tr>
<td></td>
<td>PB II 11d</td>
<td>Safety significant equipment for confinement</td>
<td>HVAC/PV system (exhaust fans and plenums duct work, HEPA filters) HVAC/PV process hood isolation damper HVACD system (exhaust HEPA filter exhaust duct work, isolation damper) Standby electrical power (diesel generator and HVAC/PV system restart circuit) HVACB isolation dampers (outside air inlets)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>PB H 11e</td>
<td>Safety significant equipment for monitoring</td>
<td>Reference air system (reference air header differential pressure alarms for process bays)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>PB L 11d</td>
<td>TSR</td>
<td>Perform process connector leak test before processing Close long axial process tube port before closing filtered process exit port at the end of processing</td>
<td></td>
</tr>
</tbody>
</table>
Table B3-16  Summary of Safety Features Required to Mitigate or Prevent a Multi-Canister Overpack Internal Hydrogen Explosion  (6 sheets)

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4.0)</th>
<th>NRC ITS*</th>
</tr>
</thead>
<tbody>
<tr>
<td>12 Hydrogen explosion within an MCO due to instrumentation failure</td>
<td>PB B 02a</td>
<td>Prevent hydrogen explosion Protect against instrumentation inaccuracy</td>
<td>Safety class equipment (performing a safety significant function) for detection SCIC process bay high temperature trip Safety class equipment (performing a safety significant function) for confinement, purge and pressurize Cask-MCO SCHe system Lines and valves to isolate and purge the MCO Safety significant equipment for confinement HVAC/PV system (exhaust fans and plenums duct work HEPA filters) HVAC/PV process hood isolation damper HVAC/C system (exhaust HEPA filter exhaust duct work, isolation damper) Standby electrical power (diesel generator and HVAC/PV system restart circuit) HVACB isolation dampers (outside air inlets) Safety significant equipment for monitoring Reference air system (reference air header differential pressure alarms)</td>
<td>B</td>
</tr>
</tbody>
</table>

* NRC ITS: B, A, B, A, B, B
### Table B3.16  Summary of Safety Features Required to Mitigate or Prevent a Multi-Unit Overpack Internal Hydrogen Explosion

<table>
<thead>
<tr>
<th>Checklist designator</th>
<th>Candidate accident</th>
<th>Safety function</th>
<th>Safety Features (described in Chapter B4.0)</th>
<th>NRC ITFS</th>
</tr>
</thead>
<tbody>
<tr>
<td>PB L 01</td>
<td>I3 Hydrogen explosion within an MCO due to lack of MCO control caused by facility fire</td>
<td>Prevent hydrogen explosion</td>
<td>SCIC process bay high temperature trip</td>
<td>B</td>
</tr>
<tr>
<td>PB L 02</td>
<td></td>
<td>Protect a processing area against external fire</td>
<td>TSR, Combustible loading limited</td>
<td></td>
</tr>
<tr>
<td>PB L 03</td>
<td></td>
<td>Process bay against other processing area</td>
<td>Restore bay temperatures following process bay high temperature trip</td>
<td></td>
</tr>
<tr>
<td>PB L 04</td>
<td></td>
<td>(administrative area)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>PB L 05</td>
<td></td>
<td>Other transfer corridor</td>
<td></td>
<td></td>
</tr>
<tr>
<td>PB L 06</td>
<td></td>
<td>Other nonprocessing area</td>
<td></td>
<td></td>
</tr>
<tr>
<td>PB L 07</td>
<td></td>
<td>Other transfer corridor</td>
<td></td>
<td></td>
</tr>
<tr>
<td>PB L 08</td>
<td></td>
<td>Other nonprocessing area</td>
<td></td>
<td></td>
</tr>
<tr>
<td>PB L 09</td>
<td></td>
<td>Other transfer corridor</td>
<td></td>
<td></td>
</tr>
<tr>
<td>PB L 10</td>
<td></td>
<td>Other nonprocessing area</td>
<td></td>
<td></td>
</tr>
<tr>
<td>PB L 11</td>
<td></td>
<td>Other transfer corridor</td>
<td></td>
<td></td>
</tr>
<tr>
<td>PB L 12</td>
<td>I4 Hydrogen explosion within an MCO due to hydride reaction</td>
<td>The hydride reaction cannot cause enough hydrogen to generate an explosion</td>
<td>None</td>
<td></td>
</tr>
<tr>
<td>PB L 13</td>
<td></td>
<td></td>
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<tr>
<td>PB L 14</td>
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<tr>
<td>PB L 15</td>
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<tr>
<td>PB L 16</td>
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<td></td>
</tr>
</tbody>
</table>

**Annex B — Cold Vacuum Drying Facility**

November 1999
<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4 0)</th>
<th>NRC ITS</th>
</tr>
</thead>
<tbody>
<tr>
<td>15 Hydrogen explosion within an MCO due to loss of support utilities</td>
<td>PB F 02a, PB F 05, SB F 01b, SB F 02b, OU P 04, OU R 02, OU R-03, OU R 04</td>
<td>Prevent hydrogen explosion, Place the MCO in a safe configuration during a loss of support utilities</td>
<td>Safety-class (performing a safety significant function) equipment for detection, SCIC process bay high temperature trip</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Safety class (performing a safety significant function) equipment for confinement, purge and pressurize</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Cask-MCO, SCHe system, Lines and valves to isolate and purge the MCO</td>
<td>A</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Safety significant equipment for confinement: HVAC/PV system (exhaust fans and plenums duct work, HEPA filters), HVAC/PV process hood isolation damper, HVACD system (exhaust HEPA filter exhaust duct work, isolation damper), Standby electrical power (diesel generator and HVAC/PV system restart circuit), HVACB isolation dampers (outside air inlets)</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Safety significant equipment for monitoring: Reference air system (reference air header differential pressure alarms for process bays)</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>TSR</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Crane movement restricted during MCO processing except as part of an approved recovery plan</td>
<td></td>
</tr>
<tr>
<td>Candidate accident</td>
<td>Checklist designator*</td>
<td>Safety function</td>
<td>Safety features (described in Chapter B4.0)</td>
<td>NRC ITS*</td>
</tr>
<tr>
<td>---------------------</td>
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</tr>
<tr>
<td>16 Hydrogen explosion within an MCO due to line break caused by a seismic event</td>
<td>PB R 01a OU R 01a</td>
<td>Prevent hydrogen explosion Place the MCO in a safe configuration during a seismic event</td>
<td>Safety class (performing a safety significant function) equipment for detection SCIC seismic trip Safety-class (performing a safety significant function) equipment for shutdown Cask—MCO SCHe system Lines and valves to isolate and purge the MCO Safety significant equipment for confinement HVAC/PV system (exhaust fans and plenums duct work, HEPA filters) HVAC/PV process hood isolation damper HVACD system (exhaust HEPA filter exhaust duct work, isolation damper) Standby electrical power (diesel generator and HVAC/PV system restart circuit) HVAC B isolation dampers (outside air inlets) Safety significant equipment for monitoring Reference air system (reference air header differential pressure alarms for process bays) TSR Trailer placement controlled such that movement from seismic events will not impact key shutdown systems</td>
<td>B A B B</td>
</tr>
</tbody>
</table>
Table B3-16  Summary of Safety Features Required to Mitigate or Prevent a  
Multi-Canister Overpack Internal Hydrogen Explosion  (6 sheets)

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator&lt;sup&gt;a&lt;/sup&gt;</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4 0)</th>
<th>NRC ITS&lt;sup&gt;b&lt;/sup&gt;</th>
</tr>
</thead>
</table>
| I 7 Hydrogen explosion within an MCO due to contamination of helium supply | PB H 06k                          | Prevent a contamination of helium supply | TSR  
Shipment paperwork verified for helium supply gas content during receipt |                     |

<sup>a</sup>Checklist designators are from HNF SD SNF HIE 004 1999 *Cold Vacuum Drying Facility Hazard Analysis* Rev 4 Fluor Daniel Hanford, Incorporated, Richland, Washington

<sup>b</sup>U.S. Nuclear Regulatory Commission important to safety classifications Category A = critical to safe operation, Category B = major impact on safety Category C = minor impact to safety

<sup>c</sup>Lines and valves to isolate the MCO include the isolation valves (and filters on air supply to valve actuators) in the VPS general service helium system, PWC system, and SCHe system

<sup>d</sup>Significant hydride reactions are not possible (HNF 2786 1998 *Assessment of Potential for Rapid Ignition of Submerged N Reactor Fuel* Fluor Daniel Hanford, Incorporated, Richland, Washington.)

HEPA = high-efficiency particulate air (filter)  
HVACB = process bay recirculation HVAC system.  
HVACC/PV = process bay local exhaust HVAC and process vent system.  
HVACD = process general supply/exhaust HVAC system.  
ITS = important to safety  
MCO = multi-canister overpack.  
NRC = U.S. Nuclear Regulatory Commission  
SCHe = safety class helium.  
SCIC = safety class instrumentation and control  
TSR = technical safety requirement.  
VPS = vacuum purge system
I 3 Hydrogen explosion within an MCO due to loss of MCO control caused by facility fire (significant air ingress into the MCO)

The accident scenarios identified in the hazard analysis for the I 3 bin represent internal hydrogen explosions caused by process upsets during fire in the bay (HNF-SD-SNF-HIE-004). The controls in the I 3 bin prevent an internal hydrogen explosion by protecting safety-related SSCs from damage during a fire in the process bay. Controlling the combustibles in the bay prevents such damage. In addition, credit is taken for controls to protect a bay temperature assumption made in the fire hazards analysis.

Safety-class equipment (performing a safety-significant function) to prevent an internal hydrogen explosion:

- SCIC process bay high temperature trip

The SCIC process bay high temperature trip acts to protect an initial condition assumption in the fire hazards analysis.

Assumptions made that require protection by TSRs are listed below:

- Combustible loadings limited

While an MCO is present in the facility, combustible loadings are limited as determined by the fire hazard analysis implementation plan (SNF-4942). These limits ensure that any fire in the CVDF does not result in uncontrolled releases (e.g., fire-caused loss of process control).

- Restore bay temperature following a process bay high temperature trip

On high bay temperature trip alarm, operations must return the process bay temperature to within acceptable limits.

I 4 Hydrogen explosion within an MCO due to hydride reaction

The accident scenarios identified in the hazard analysis for the I 4 bin represent internal hydrogen explosions caused by hydride reactions (HNF-SD-SNF-HIE-004). The uranium hydride reaction, in and of itself, has been shown to produce insufficient quantities of hydrogen to cause an internal hydrogen explosion. Thus, no controls are necessary for this bin. Hydrides have been accounted for as a contributor to other internal hydrogen explosion bins and other DBAs by an increase in the effective corrosion reaction rate.

No additional requirements result from analysis of this accident.
15 Hydrogen explosion within an MCO due to loss of support utilities (significant air ingress)

The accident scenarios identified in the hazard analysis for the 15 bin represent internal hydrogen explosions caused by loss of support systems resulting from accidents in adjacent bays, crane load drops, and accidents in the spare bay. This bin also includes loss-of-power events caused by external forces (e.g., vehicle accident), flooding, and lightning strike. The controls in the 15 bin prevent an internal hydrogen explosion by protecting against the malfunction of SSCs and by controlling specific initiators that would result in a loss of support utilities. For an internal explosion to be possible, a line leak must have occurred to allow air ingress. To mitigate the gaseous release associated with a line leak, credit is taken for confinement systems to control that release. Note that the confinement systems are not credited with mitigation of the internal explosion release but only with mitigation of the gaseous release associated with the line leak.

Assumptions made that require protection by TSRs are listed below:

- Crane movement restricted during MCO processing except as part of an approved recovery procedure

  Bridge crane movement is not allowed from the time that draining operations begin until the proof-of-dryness demonstration is complete and the MCO isolation valves are closed, except as part of an approved recovery procedure. This restriction ensures that key process lines cannot be sheared by crane movement, which could result in damage to systems relied upon to preclude an internal hydrogen explosion.

16 Hydrogen explosion within an MCO due to line break caused by a seismic event

The accident scenarios identified in the hazard analysis for the 16 bin represent internal hydrogen explosions caused by process upsets following a seismic event (HNF-SD-SNF-HIE-004). The controls in the 16 bin prevent an internal hydrogen explosion by removing reliance on nonqualified SSCs (piping and valves) whose failure could allow air ingress into the MCO. The MCO will be vented and confinement SSCs are credited with controlling the potential gaseous release. Note that the confinement systems are not credited with mitigation of the internal explosion release but only with mitigation of the gaseous release.

Safety-class (performing a safety-significant function) equipment for detection

- SCIC seismic trip

  On detecting a seismic event, the SCIC system seismic trip isolates the MCO and actuates the SCHe system and de-energizes the tempered water (annulus) system heater to ensure the seismic event will not initiate thermal runaway.
Assumptions made that require protection by TSRs are listed below

- Trailer placement controlled such that movement from seismic events will not impact key shutdown systems

Before connecting CVDF systems to the MCO, the cask, which is located upon the transporter, must be positioned such that the central axis of the cask is no greater than an established safe distance from the ideal horizontal placement, as identified in Chapter B4. This control protects the MCO during seismic events

17 Hydrogen explosion within an MCO due to contamination of helium supply

The accident scenarios identified in the hazard analysis for the I7 bin represent internal hydrogen explosions caused by fuel reactions with contaminants in the helium supply (HNF-SD-SNF-HIE-004) The controls in the I7 bin prevent an internal hydrogen explosion by controlling the helium supply sent to the MCO

Assumptions made that require protection by TSRs are listed below

- Shipment paperwork verified for helium supply gas content during receipt

The manufacturer's paperwork and shipping papers should be checked upon helium cylinder receipt at the CVDF, both the normal and safety-class supply, to verify that the cylinder's contents were sampled by the supplier and that the sample met the required purity specification of > 99% helium

B3 4.2.5 Multi-Canister Overpack Thermal Runaway Reaction Thermal runaway reaction accidents would occur if heat removal from the MCO is not sufficient to prevent fuel elements and scrap fuel from heating up as a result of decay heat and chemical reaction heat. As the temperature increases, the chemical reaction rate increases, producing more gas and heat. If the thermal runaway reaction is allowed to continue, the temperature would continue to increase until all fuel or water is consumed by the fuel–water chemical reaction. Detailed calculations of the consequences of an MCO thermal runaway reaction are provided in SNF-2770, Chapter 6. The SSCs designated to prevent the MCO runaway reaction are designated safety class because unmitigated consequences exceed the offsite release limit of 0.5 rem (0.005 Sv).

Version 1.3.2 of the HANSF computer code (SNF-3650) was used to simulate the thermal runaway cases. HANSF calculates the fuel, gas, and MCO structure temperatures and MCO pressure based on uranium-water and hydride-water reactions, decay heat, and heat transfer. The code also calculates the particulate or MAR for the thermal cases over time. Validation and verification assurance for the HANSF code at the Hanford Site is described in SNF-5226.

*Comparison Cases Simulated with HANSF 1.3.2 to Supplement Thermal Analyses Documented in HNF-SD-SNF-CN-023*
Safety-class features that prevent a thermal runaway accident include portions of the tempered water system to provide adequate heat removal, a tempered water high temperature trip, the SCIC and associated safety-class instruments to detect process upsets and activate the SCHe system, and the SCHe system to provide an MCO purge and vent function. These safety-class preventive features reduce the estimated thermal runaway accident frequency into the beyond extremely unlikely frequency range (less than $10^{-6}$ per year). No additional mitigation features are required.

**B3 4 2 5 1 Scenario Development** The CVDF hazard analysis report (HNF-SD-SNF-H1E-004) identified several potential accidents leading to a thermal reaction runaway. These included events such as loss of annulus water, degraded vacuum pump capability, and high tempered water temperature. From the identified potential thermal runaway reaction accidents, a bounding accident of a loss of water in the annulus between the MCO and transportation cask was selected and developed as the DBA. This accident was selected as the DBA based on a qualitative estimate of the likelihood of the binned thermal runaway events.

All of the thermal runaway scenarios require an insufficient rate of heat removal from the inside of the MCO, which is considered to be a process parameter upset. The amount of heat removal capability needed depends on the amount of decay heat and the fuel–water reaction rate in the MCO. Under normal conditions, the main heat sink for the MCO is the flowing annulus water. Because of the high thermal conductivity of water, heat removal from the MCO is still adequate under conditions of stationary annulus water and off-center MCO in cask. Any metal-to-metal contact would improve thermal conductivity. Higher temperature annulus water has less heat removal capability than cooler temperature annulus water. The temperature capability of the tempered water system is from 10 °C to 100 °C (50 °F to 212 °F) with a nominal operating value of from 40 °C to 50 °C (122 °F). If the annulus water is not flowing or is lost, the air surrounding the cask becomes the heat sink. The temperature of the annulus water is limited to 50 °C to ensure that the process remains within analyzed bounds while maintaining a temperature high enough to dry the fuel. The 50 °C limit also ensures that the fuel and MCO do not experience elevated temperatures such as postulated in the unmitigated scenario. Because of conservatisms built into the analysis, a short duration temperature above 50 °C could be tolerated provided that the tempered water heater is deactivated at ≤ 50 °C.

For an MCO with insufficient heat removal, the temperatures in the MCO essentially continue to increase until the water is consumed. With sufficient initial water available, temperatures in excess of the eutectic temperature for uranium and iron (725 °C) (HNF-SD-SNF-SARR-005) could be reached, with the potential for localized damage to MCO internal components (e.g., scrap and fuel baskets).

With the postulated loss of cask–MCO annulus water (e.g., leak or failure of lower port connection on cask), heat generation inside the MCO exceeds the heat removal capability. In the DBA scenario, there is no helium flow through the MCO, but there is pressure relief through the normal processing vent path. The loss of annulus water causes the temperatures inside the MCO to slowly increase at first, but after 10 hours, the rate of temperature rise increases significantly.
About 11.5 hours after loss of annulus water, some MCO fuel temperatures exceed 725 °C but no fuel temperatures exceed 1,010 °C, which is below the uranium melting temperature of 1,125 °C.

The high fuel temperature results in the continuous release of particulate from within the MCO, driven by the gases generated from the uranium fuel reaction within the hot MCO. This release would continue as long as unreacted uranium is in contact with water vapor. The loss of annulus water has been shown to be credible for the unmitigated accident scenario (see SNF-2770, Appendix A).

The fuel temperatures reached in localized regions inside the MCO could exceed the uranium-steel eutectic temperature and allow the formation of some eutectic. This could cause spalling and limited structural damage to MCO components such as the scrap and fuel baskets. The thermal calculations (SNF-2770) with the HANSF code (SNF-3650) predict that about 700 kg of fuel in a Mark IV MCO (all in the bottom scrap basket) will reach a temperature in excess of 700 °C, with a maximum center post temperature less than 1,000 °C. The maximum wall temperature for the Mark IV MCO during the loss of annulus water is 500 °C. For the Mark IA MCO, no fuel reaches temperatures in excess of 710 °C, with a maximum center post temperature of 640 °C. The maximum Mark IA MCO wall temperature is less than 520 °C. These calculations show that no temperatures are reached during the thermal runaway sufficient to cause structural damage to the MCO walls for either fuel type or to the Mark IA baskets and center post. Only limited damage to the bottom Mark IV scrap basket and center post is predicted. Thus, the safety-class confinement and criticality contingency function of the MCO and the criticality contingency function of the Mark IA basket center post is not challenged during the thermal runaway. Because the Mark IV baskets are not safety class, limited structural damage would be tolerable.

The water in the MCO reacts with exposed fuel, causing a continuous stream of gases and entrained particulate to flow out of the MCO. Both the scrap fuel and exposed fuel elements react with the water vapor that is flowing through the MCO. The water vapor continues to react with uranium and uranium hydride until the water is completely consumed, which for the scrap baskets, is about 14 hours after the loss of annulus water (SNF-2770).

The bounding unmitigated thermal runaway scenario, therefore, requires the following conditions:

- MCO processing without helium flow (the flow of helium does not prevent the thermal runaway but lengthens the time required to reach high temperatures).

- Loss of cask-MCO annulus water.

- Continuous release through a failed process line (the release continues as long as unreacted uranium is in contact with water vapor or oxygen).
The following assumptions are used in the analyses

- The MCO thermal runaway reaction scenario is assumed to begin when the water draining process is finished but some undrained (residual) water is still present in the MCO to react with the fuel. It is assumed that 26.5 kg of free undrained (residual) water (15 kg in the top scrap basket, 6 kg in the bottom scrap basket, 18 kg in three fuel baskets, and 1 kg in the gap between the bottom fuel basket and the MCO bottom plate) is present when the thermal runaway accident is initiated (SNF-2770).

- Uranium hydrate contains 119 kg of water (HNF-SD-SNF-TI-015).

- The MCO is at bounding configuration with two scrap baskets:
  - Enhancement factor or rate multiplier fuel–water reaction equals 10 (HNF-SD-SNF-TI-015).
  - Surface area equals 12 m² (HNF-SD-SNF-TI-015).
  - Decay heat rate equals 705 W for three fuel baskets and two scrap baskets (HNF-SD-SNF-TI-015).

- Uranium hydride is included using an enhancement factor or rate multiplier of 12 to account for the mass fraction of hydride and its surface area (HNF-SD-SNF-TI-015). This factor is multiplied by the reaction rate from literature in order to conservatively enhance the oxidation rate. The rate multipliers of 12 for hydride–water reactions and 10 for uranium–water reactions are kept separate in the model because each reaction produces different amounts of heat and hydrogen. Hence, the rate multiplier of 22 is just a convenient way to say that both reactions are modeled.

- The effects of fuel crumbling at high temperatures were not explicitly included in the analysis but were included in the rate multiplier of 22. Increased reaction rates were not observed experimentally in fuel samples that crumbled over time (HNF-4206), although other samples that did not crumble had lower rates by a factor of 2 to 4. The analysis assumes that the reaction rate multiplier of 10 for uranium and 12 for uranium hydride for a combined rate multiplier of 22 bounds any effects of increased surface area caused by fuel crumbling. For comparison, the crumbled samples had rate multipliers less than 4 (HNF-4206).

- No hydrogen gettering takes place because of the presence of water at the beginning of the scenario and air at the end of the scenario. Hydrogen gettering or uranium–hydrogen reaction (\( \text{U} + 1.5 \text{H}_2 \rightarrow \text{UH}_3 \)) has been shown not to occur to any significant degree in the presence of water or oxygen (SNF-3650).
About 15 kg of particulate are generated between the time the MCO leaves the K Basins and the time it is drained at the CVDF. This newly generated particulate is available to be entrained by exiting gases during a release.

With the high temperatures produced by the thermal runaway conditions, the uranium hydrates will decompose and produce more water (up to 1.19 kg) for the fuel-water reactions.

The water contained in aluminum hydroxide, a maximum of about 3.3 kg (HNF-SD-SNF-TI-015), is used in this analysis to account for aluminum hydroxide decomposition at elevated temperatures. Only three baskets have temperatures in excess of 150 °C where some aluminum hydroxide could decompose (ALCOA 1987, Figure 4-4). A total of about 0.74 kg of hydroxide water is released for this scenario.

**B3 4 2 5 2 Source Term Analysis** The respirable source term is based on the MAR (material at risk), the ARF (airborne release fraction), and the RF (respirable fraction). The MAR is the sum of the amount of particulate created before the release and the amount of particulate created during the release.

- **MAR₁** = 15 kg UO₂ generated between the time the fuel is cleaned at the K Basins and completion of the draining process. This equates to 13.2 kg of uranium (HNF-SD-SNF-TI-015). This value was not calculated by the HANSF code and is only used as part of the following MARs.

- **MAR₂** = 52 kg UO₂ for the 12-hour continuous release. This equates to about 46 kg of uranium (includes MAR₁). The amount of particulate in MAR₂ that exceeds that in MAR₁ is generated by the continuing corrosion reaction under the DBA conditions (SNF-5226). This newly formed particulate is assumed to be susceptible to release in addition to MAR₁.

- **MAR₃** = 178 kg UO₂ for the 24-hour continuous release. This equates to about 157 kg of uranium (includes MAR₁). The amount of particulate in MAR₃ that exceeds the amount in MAR₁ is generated as a result of the continuing corrosion reaction (SNF-5226).
Airborne Release Fraction and Respirable Fraction  The ARF of all the radionuclides in the fuel, such as the fission products (e.g., cesium), may not be the same as the uranium particulate release under very high temperature conditions. For example, the cesium in the fuel could vaporize at temperatures in excess of 600 °C, increasing the rate at which it becomes airborne (DOE-STD-1027-92). However, cesium and the other nontransuranic radionuclides contribute less than 0.5% of the total rem per gram of SNF fuel dose factor of the SNF (HNF-SD-SNF-TI-059).

The bounding value for the respirable release fraction for uranium at temperatures greater than 500 °C is $1 \times 10^3$ (DOE-HDBK-3010-94, page 4-3). The bounding respirable release fraction for falling molten droplets at temperatures greater than 1,100 °C is $1 \times 10^2$ (DOE-HDBK-3010-94, page 4-3). Since falling molten droplets with high surrounding gas velocities are not the physical situation in the high-temperature MCO, $1 \times 10^2$ is considered overly conservative. However, since some fuel temperatures exceed 1,000 °C, and may approach the melting temperature of uranium (1,125 °C), the $1 \times 10^3$ release fraction for fuel temperature greater than 500 °C may not be conservative enough. Hence, an intermediate respirable release fraction of $5 \times 10^3$ was selected for the analysis.

Additional analyses using the HANSF version 1.3.2 code (SNF-3650) may adequately demonstrate that fuel melting does not occur. The implications of a revised thermal runaway DBA with reduced consequences (as a result of not incorporating a molten fuel ARF) are being evaluated further to assess impact on selected controls. The ARF used in this analysis is conservative.

The source term $M_2$, for the 12-hour continuous release is calculated by multiplying $\text{MAR}_2$, the particulate MAR for the 12-hour continuous release, by the product of ARF and RF and converting to uranium metal as follows:

$$M_2 = (\text{MAR}_2)(\text{ARF} \times \text{RF})$$

$$= (46 \text{ kg U})(1,000 \text{ g/kg})(5 \times 10^3)$$

$$< 230 \text{ g U}$$

The source term $M_3$, for the 24-hour continuous release (offsite) is calculated by multiplying $\text{MAR}_3$, the particulate MAR for the 24-hour continuous release, by the product of ARF and RF and converting to uranium metal as follows:

$$M_3 = (\text{MAR}_3)(\text{ARF} \times \text{RF})$$

$$= (157 \text{ kg U})(1,000 \text{ g/kg})(5 \times 10^3)$$

$$< 785 \text{ g U}$$
B3 4 2 5 3 Consequence Analysis  The radiological dose is calculated using the following equation

\[ DE = M \times \frac{\chi}{Q'} \times BR \times UD \]

where

- \( DE \) = effective dose equivalent based on inhalation exposure only (rem)
- \( M \) = respirable quantity released into the air (grams)
  - \( M_2 \) for 12-hour (onsite) continuous release (225 g U)
  - \( M_3 \) for 24-hour (offsite) continuous release (785 g U)
- \( \frac{\chi}{Q'} \) = air transport factor (s/m³)
- \( BR \) = average inhalation rate during the blowdown and 12-hour release (m³/s)
- \( UD \) = committed effective dose equivalent per unit gram inhaled

Unmitigated Consequences  The dose calculation equation is used to calculate the dose to the offsite receptor

\[ D_{\text{offsite}} = M_3 \times \frac{\chi}{Q'} \times BR \times UD \]

\[ = (785 \text{ g U})(6.50 \times 10^{-6} \text{ s/m}^3)(2.64 \times 10^{-4} \text{ m}^3/\text{s})(4.38 \times 10^5 \text{ rem/g U}) \]

\[ = 0.590 \text{ rem (0.0059 Sv)} \text{ for 24-hour continuous release} \]

The dose consequences at the remaining receptor sites are calculated in the same manner and are shown in Table B3-17 for the continuous release.

Mitigated Consequences  With safety-class features credited for this event the accident is prevented. The key safety-class features that combine to prevent the thermal runaway reaction are (1) the tempered water (annulus) system which removes heat from the MCO and fuel, and (2) the manual capability to provide annulus water and (3) the SCHe system, which provides pressure vent and cools the fuel, reducing the fuel-water reaction rate. Credit also is taken for restriction of crane movement during processing to protect the tempered water (annulus) system and the other process lines. With the credited preventive features the frequency of the thermal runaway DBA is beyond extremely unlikely so release limits are met. In addition credit is taken for local exhaust operation to filter and dilute SCHe flows (local exhaust is not credited except as it filters and dilutes the SCHe flows).
## Table B3-17 Dose Consequence Summary for High-Temperature Thermal Runaway Reaction

<table>
<thead>
<tr>
<th>Receptor location (distance, direction)</th>
<th>Duration (hours)</th>
<th>$\chi/Q$ a (s/m³) (without stack or plume meander)</th>
<th>Unmitigated dose b (rem (Sv))</th>
<th>Unmitigated evaluation guideline/c release limit (Sv)</th>
<th>Mitigated dose (Sv)</th>
<th>Mitigated evaluation guideline/release limit (Sv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Onsite (100 m E)</td>
<td>12 (continuous)</td>
<td>6.28 E 03</td>
<td>210 (2 1)</td>
<td>10 (1 0 E-01)</td>
<td>Prevented e</td>
<td>Not applicable, accident prevented</td>
</tr>
<tr>
<td>Columbia River (650 m NW)</td>
<td>12 (continuous)</td>
<td>1 99 E 04</td>
<td>6 7 (0 067)</td>
<td></td>
<td>Prevented e</td>
<td>--</td>
</tr>
<tr>
<td>100 Area Fire Station (3 750 m ESE)</td>
<td>12 (continuous)</td>
<td>2 73 E-05</td>
<td>0 92 (0 0092)</td>
<td>Prevented e</td>
<td>--</td>
<td>--</td>
</tr>
<tr>
<td>Hanford Site boundary (10,090 m W)</td>
<td>24 (continuous)</td>
<td>6 50 E 06</td>
<td>0 59 (0 0059)</td>
<td>5 0 (0 05)</td>
<td>Prevented</td>
<td>Not applicable, accident prevented</td>
</tr>
</tbody>
</table>

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a Fifty year committed effective dose equivalent.

b Evaluation guideline (for onsite receptor) and offsite release limit for unmitigated accidents in the unlikely frequency category (i.e. frequency between $10^4$ and $10^7$ per year).

c Mitigated accident frequency of beyond extremely unlikely

Failure of preventive features is beyond extremely unlikely ($< 10^6$ per year).
Prevention is achieved through a tempered water (annulus) system design that provides a high level of assurance that annulus water will not be lost, thus sufficient MCO heat removal is maintained. In addition, detection and alarm of the loss of annulus water levels ensures operator actions for restoration. The safe and stable state following process upsets is for the MCO to have pressure relief, a helium cover inside, annulus water at or below 50 °C available (not necessarily flowing) for heat transfer purposes, and process bay temperatures less than 115 °F.

The safety-class function of the tempered water (annulus) system is to remove heat from the MCO, thus preventing excessive fuel reactions. The key requirement of the annulus water is that it should be at a high level in the cask-MCO annulus so that good thermal conductivity exists between the MCO and cask, which will help cool the MCO and prevent a thermal runaway reaction accident. Thermal analyses performed in HNF-SD-SNF-CN-023, *Thermal Analysis of Cold Vacuum Drying of Spent Nuclear Fuel* (1998), demonstrate the capability of the tempered water (annulus) system to provide the necessary thermal conductivity when annulus water levels are high. The design provides the capability for operator actions to restore low annulus water levels. The temperature of the annulus water also is important. The thermal analysis assumes that annulus water does not exceed 50 °C (122 °F). A safety-class tempered water high temperature trip is provided to prevent annulus water overheating. The thermal analysis also assumes that the temperature in the process bay (the final heat sink if the tempered water is not flowing) is less than 115 °F. A high-bay-temperature trip initiates a safety-class alarm (the isolation and purge alarm) in the control room to alert operators so they can restore bay temperatures to normal.

**B3 4 2 5 4 Comparison to Guidelines**

**Comparison of Unmitigated Doses.** The unmitigated frequency of the event tree sequences that represent this DBA, including an unfiltered release, is $5 \times 10^{-4}$ per year, thus this unmitigated DBA is unlikely (see SNF-2770, Appendix A). The unmitigated sequence considered the processing of 200 MCOs per year and failure of single-walled tempered water system inlet piping. The unmitigated radiological offsite dose for this event is above the safety-class dose limit of 0.5 rem, but below the offsite release limit for an unlikely event of 5.0 rem.

**Comparison of Mitigated Doses.** The mitigated frequency of the event tree sequences that represent this DBA as a thermal runaway excursion is $1 \times 10^{-5}$ per year without release. Thus this mitigated DBA is beyond extremely unlikely (see SNF-2770, Appendix A). The accident is prevented with safety-class features (i.e., tempered water double-walled pipe and the ability to detect low annulus water and refill the system). Failure of prevention features is beyond extremely unlikely. Thus, the offsite release limits and onsite risk evaluation guidelines are met.
B3 4 2 5 5 Summary of Safety-Class Structures, Systems, and Components and Technical Safety Requirement Controls

Under normal operating conditions, annulus water levels and helium in the MCO provide adequate heat transfer to prevent thermal runaway reactions. Under upset or accident conditions, safety-class equipment is required in order to ensure this capability is not lost. The heat transfer from the MCO could be compromised by a loss of water level or high tempered water temperatures. The heat transfer within the MCO could be compromised by a loss of helium.

The checklist designators included in the accident bins, other than the accident selected as the DBA, represent additional accident sequences slightly different than the DBA. All of these binned accidents are bounded by the DBA because they have lesser or equivalent worst-case consequences and frequencies.

The accident scenarios identified in the hazard analysis for the T1 bin represent thermal runaway accidents caused by failure of the proof-of-dryness demonstration (instrumentation failure or operator error) and degraded vacuum pump and subsequent possible elevated fuel temperatures. This bin also includes thermal runaways in a loss-of-power event caused by external forces (e.g., vehicle accident), flooding, and lightning strikes (HNF-SD-SNF-HIE-004). The controls in the T1 bin prevent a thermal runaway by ensuring that an adequate heat sink is available. In addition, credit is taken for controls that prevent bypass of safety features by incorrectly proceeding with the cold vacuum drying process or by operator error.

The safety-class equipment designated to prevent the dose consequences of thermal runaway reaction accidents are described below.

Safety-class equipment for detection of process upset

- Tempered water (annulus) system level alarm

  Redundant liquid level indicators are used by the tempered water (annulus) system to detect a low water level in the cask-MCO annulus and to provide a signal to the SCIC system to actuate the low-level alarm.

- Tempered water (annulus) system level check petcocks

  The level check petcocks provide a manual means of verifying the presence of water in the annulus. The petcocks are located in and above the double-walled portion of the tempered water (annulus) piping and are connected to the outer and inner pipes above the height of the fuel in the MCO. If a low-level alarm is received, the petcocks can be used to verify the presence of water in the inner or outer pipe, thus indicating the presence of water in the annulus.
Safety-class equipment for heat removal

- Tempered water (annulus) system piping and antisiphon valves

  A portion of the tempered water (annulus) system contains double-walled safety-class piping and antisiphon valves to ensure retention of a minimum water level above the elevation of the SNF inside the MCO

- Manual refill piping and vent port

  The tempered water (annulus system) provides manual refill capability, which could be required if a low annulus water level were detected

- Cask-MCO

  The cask-MCO is a major part of the pressure boundary for tempered water (annulus) and a confinement of radioactive materials during processing and provides connections for the process piping to the SCHe system

Assumptions made that require protection by TSRs are listed below

- Administrative procedure for restricting crane movement during MCO processing except as part of an approved recovery procedure

  Bridge crane movement is not allowed from the time that draining operations begin until the proof-of-dryness demonstration is complete and the MCO isolation valves are closed except as part of an approved recovery procedure. This restriction ensures that key process lines cannot be sheared by crane movement which could result in loss of annulus water if the lower tempered water (annulus) line were damaged

- Procedure to verify the results of the pressure rebound tests before continuing process steps

  An initial pressure rebound test surveillance (pressure rise test) must be met before entry into the proof-of-dryness demonstration is allowed. Similarly, a proof-of-dryness demonstration surveillance must be met before the final pressure rebound test steps can begin. Finally, a final pressure rebound test must be met before shipment preparation steps can begin
Close long axial process tube port before closing filtered process exit port at the end of processing.

When isolating the MCO at the end of processing, the long axial process tube port must be closed at least 5 minutes before closing the filtered process exit port. This ensures that an SCIC trip will occur if the procedure to isolate the MCO is inadvertently conducted on an MCO still to be processed.

The bounding thermal runaway accident (T1) and the other accidents identified in the CVDF hazard analysis report (HNF-SD-SNF-HIE-004) that can potentially involve a thermal runaway reaction in an MCO are itemized in Table B3-18, along with corresponding checklist designators from the hazard analysis report, safety functions, and SSCs.

The accidents within the remaining bins require the following safety SSCs and TSRs, in addition to those identified for the DBA (T1).

T2 Thermal runaway reaction in MCO due to instrumentation failure.

The accident scenarios identified in the hazard analysis for the T2 bin represent thermal runaway accidents caused by elevated fuel temperatures resulting from a high bay temperature and high annulus water temperature (HNF-SD-SNF-HIE-004). The controls in the T2 bin prevent a thermal runaway by monitoring key process parameters and by protecting safety-related SSCs from damage. SCHe actuation is credited, therefore confinement and dilution provided by the local exhaust system also are credited.

Safety-class equipment for detection of process upset or instrumentation failure:

- VPS instrumentation (pressure)
  
  Pressure indicators on the VPS provide MCO internal pressure information to the SCIC system, which initiates MCO isolation and SCHe actuation during process upset conditions.

- SCIC vacuum cycle timer (8-4-4 logic)
  
  The SCIC system uses programmable logic controllers, wiring to process instrumentation, vacuum limit timer, signals from seismic detectors and temperature monitors, system controls, and output relays to perform an MCO isolation and purge trip and a tempered water trip.
<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4.0)</th>
<th>NRC ITs³</th>
</tr>
</thead>
<tbody>
<tr>
<td>T 1 Runaway reaction due to loss of process control parameter (including presence of heat sink) (bounding accident, PB B 13c)</td>
<td>PB B 13c, PB B 13d, PB H 08, OU P 04, OU R 02, OU R 03, OU R 04</td>
<td>Prevent runaway reaction, Maintain parameters within limits</td>
<td>Safety class equipment for detection, Tempered water (annulus) system level alarm, Tempered water (annulus) system level check petcocks, Safety class equipment for heat removal, Tempered water (annulus) system piping and antisiphon valves, Manual refill piping and vent port, Cask-MCO, TSR, Administrative procedure for restricting crane movement during MCO processing except as part of an approved recovery procedure, Close long axial process tube port before closing filtered process exit port at the end of processing, Procedure to verify the results of the pressure rebound tests before continuing process steps, Defense in depth equipment for confinement, purge and pressurize, General-service helium system safety-class flow instrumentation, SCHé system, Lines and valves to isolate and purge the MCO</td>
<td>B, BB, BA</td>
</tr>
</tbody>
</table>
Table B3-18 Summary of Safety Features Required to Mitigate or Prevent a Thermal Runaway Reaction  (6 sheets)

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4 0)</th>
<th>NRC ITS$^b$</th>
</tr>
</thead>
<tbody>
<tr>
<td>T 2 Runaway reaction in MCO due to instrumentation failure</td>
<td>PB B 02b PB B 03a</td>
<td>Protect instrumentation operability and analysis assumption</td>
<td>Safety class equipment for detection VPS instrumentation (pressure) SCIC vacuum cycle timer (8 4 4 logic) SCIC process bay high temperature trip Tempered water (annulus) system temperature trip</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Safety class equipment for confinement, purge and pressurization Cask-MCO SCHe system Lines and valves to isolate and purge the MCO</td>
<td>A B B</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Safety significant equipment for dilution and filtration to support SCHe operation HVAC/PV system (exhaust fans and plenums duct work, HEPA filters) HVAC/PV process hood isolation damper Standby electrical power (diesel generator and HVAC/PV system restart circuit)</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>TSR Administrative procedure for restricting crane movement during MCO processing except as part of an approved recovery procedure Restore bay temperatures following process bay high temperature trip</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Defense in depth HVAC maintains operating bay temperature</td>
<td></td>
</tr>
</tbody>
</table>

Annex B — Cold Vacuum Drying Facility
<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator*</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4.0)</th>
<th>NRC ITS(^b)</th>
</tr>
</thead>
<tbody>
<tr>
<td>T 3 Runaway reaction in MCO due to loss of MCO control caused by facility fire</td>
<td>PB L 01, PB L 02, PB L 03, PB L 04, PB L 05, PB L 06, PB L 07, PB L 08, PB L 09, PB L 10, PB L 13, PB L 14, PB L 15, PB L 16, PB P 02, OU P 02a</td>
<td>Protect a processing process bay against external fire (administrative area, transfer corridor other nonprocessing process bay) and limit the fire risk inside a processing process bay</td>
<td>Safety class equipment (performing a safety significant function) for detection. SCIC process bay high temperature trip TSR Combustible loadings limited Restore bay temperatures following process bay high temperature trip Defense in depth equipment for process upset conditions and process bay temperature detection VPS instrumentation (pressure) SCIC vacuum cycle timer (8 4 4 logic) Tempered water (annulus) system temperature trip Tempered water (annulus) system level alarm General service helium system safety-class flow instrumentation Defense in depth equipment for heat removal Tempered water (annulus) system piping and antisiphon valves Manual refill piping and vent port Cask-MCO Defense in depth equipment for shutdown SCHe system Lines and valves to isolate and purge the MCO Defense in depth Procedure to limit combustible loading Fire protection system present in each bay SCHe system Lines and valves to isolate and purge the MCO(^b)</td>
<td>B</td>
</tr>
<tr>
<td>Candidate accident</td>
<td>Checklist designtor*</td>
<td>Safety function</td>
<td>Safety features (described in Chapter B4 0)</td>
<td>NRC ITS*</td>
</tr>
<tr>
<td>-------------------------------------------</td>
<td>----------------------</td>
<td>----------------------------------------------</td>
<td>-------------------------------------------------------------------------------------------------------------</td>
<td>----------</td>
</tr>
<tr>
<td>T 4 Runaway reaction due to loss of support utilities</td>
<td>PB F 02a</td>
<td>Prevent runaway reaction</td>
<td>Safety class equipment for detection&lt;br&gt;VPS instrumentation (pressure)&lt;br&gt;SCIC (pressure/vacuum cycle timer) (8 4 4 logic)&lt;br&gt;SCIC process bay high temperature trip</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td>PB F 05</td>
<td></td>
<td>Safety class equipment for heat removal&lt;br&gt;Tempered water (annulus) system piping and antisiphon valves&lt;br&gt;Manual refill piping and vent port&lt;br&gt;Cask-MCO</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td>PB H 11f</td>
<td></td>
<td>Safety class equipment for confinement, purge and pressurization&lt;br&gt;SCHe system&lt;br&gt;Lines and valves to isolate and purge the MCO</td>
<td>A</td>
</tr>
<tr>
<td></td>
<td>SB F 01b</td>
<td></td>
<td>Safety class equipment for prevention of process upset&lt;br&gt;Deminized water and PWC drain line isolation valves interlocked closed after being used in the process</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td>SB F 02b</td>
<td></td>
<td>Safety significant equipment for dilution and filtration to support SCHe operation&lt;br&gt;HVACC/PV system (exhaust fans and plenums duct work, HEPA filters)&lt;br&gt;HVACC/PV process hood isolation damper&lt;br&gt;Standby electrical power (diesel generator and HVACC/PV system restart circuit)</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>TSR&lt;br&gt;Administrative procedure for restricting crane movement during MCO processing except as part of an approved recovery procedure</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>Defense in depth equipment for confinement, purge and pressurize&lt;br&gt;General service helium system safety class flow instrumentation</td>
<td></td>
</tr>
</tbody>
</table>
Table B3-18  Summary of Safety Features Required to Mitigate or Prevent a Thermal Runaway Reaction  (6 sheets)

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator*</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4 0)</th>
<th>NRC ITS®</th>
</tr>
</thead>
<tbody>
<tr>
<td>T 4 Runaway reaction due to loss of support utilities (continued)</td>
<td></td>
<td></td>
<td>Defense in depth equipment for MCO process testing</td>
<td></td>
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<td></td>
<td></td>
<td></td>
<td>Initial pressure rebound test prior to proof of dryness demonstration</td>
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<td></td>
<td></td>
<td></td>
<td>(temperature indicator on the tempered water [annulus] system and pressure on the VPS)</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>Final pressure rebound test following the proof of dryness demonstration after all potential water sources are isolated (temperature indicator on the tempered water [annulus] system and pressure on the VPS)</td>
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</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Defense in depth  General service helium system safety class flow instrumentation</td>
<td></td>
</tr>
<tr>
<td>T 5 Runaway reaction due to a contamination of the helium supply</td>
<td>PB H 06k</td>
<td>Prevent using a contaminated helium supply</td>
<td>TSR</td>
<td></td>
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<td></td>
<td></td>
<td></td>
<td>Shipment paperwork verified for helium receipt</td>
<td></td>
</tr>
<tr>
<td>T 6 Runaway reaction due to line break caused by a seismic event</td>
<td>PB R 01a OU R-01a</td>
<td>Prevent the runaway reaction Place the MCO is in a safe configuration following a seismic event</td>
<td>Safety-class equipment for tempered water annulus level</td>
<td></td>
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<td></td>
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<td></td>
<td>Tempered water (annulus) system piping and antisuphon valves</td>
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<td>Refill piping and vent port</td>
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<td></td>
<td>Safety equipment for detection of seismic event</td>
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<td></td>
<td>SCIC seismic trip</td>
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<td></td>
<td></td>
<td></td>
<td>Safety class equipment for confinement, purge and pressurization</td>
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<td></td>
<td></td>
<td></td>
<td>Cask-MCO  SCHy system</td>
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<td></td>
<td></td>
<td></td>
<td>Lines and valves to isolate and purge the MCO*</td>
<td></td>
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<td></td>
<td></td>
<td></td>
<td>TSR</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Trailer placement controlled such that movement from seismic event will not impact key shutdown systems</td>
<td></td>
</tr>
<tr>
<td>Candidate accident</td>
<td>Checklist designator*</td>
<td>Safety function</td>
<td>Safety features (described in Chapter B4 0)</td>
<td>NRC ITS²</td>
</tr>
<tr>
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<td>---------------------------------------------</td>
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</tr>
</tbody>
</table>

*Checklist designators are from HNF SD SNF HIE 004 1999 Cold Vacuum Drying Facility Hazard Analysis Rev 4 Fluor Daniel Hanford, Incorporated, Richland Washington

²U.S. Nuclear Regulatory Commission important to safety classifications Category = critical to safe operation, Category B = major impact on safety Category C = minor impact to safety

Lines and valves to isolate the MCO include the isolation valves (and filters on air supply to valve actuators) in the VPS general service helium system, PWC system, and SCHe system.

HEPA = high efficiency particulate air (filter)
HVAC = heating, ventilation, and air conditioning
HVACC/PV = process bay local exhaust HVAC and process vent system
ITS = important to safety
MCO = multi-canister overpack.
NRC = U.S. Nuclear Regulatory Commission
PWC = process water conditioning
SCHe = safety class helium
SCIC = safety-class instrumentation and control
TSR = technical safety requirement.
VPS = vacuum purge system
8 4 4 = 8 hour initial vacuum cycle 4 hour subsequent vacuum cycles 4 hour return to pressure between vacuum cycles
Tempered water (annulus) system temperature trip

The system has temperature indicators and transmitters to detect high tempered water (annulus) water temperature and signals the SCIC system to actuate the tempered water trip.

SCIC process bay high temperature trip

The SCIC high bay temperature trip isolates the MCO and actuates the SCHe system. A safety-class alarm alerts operations to the bay condition.

Safety-class equipment for confinement, purge, and pressurization

Cask–MCO

The cask–MCO is a major part of the pressure boundary for confinement of radioactive materials during processing and provides connections for the process piping to the SCHe system.

SCHe system

The SCHe system provides two redundant and independent paths for purging and pressurizing the MCO and venting to the process vent. The SCHe purging and pressurizing functions help to prevent elevated temperatures inside the MCO.

Lines and valves to isolate and purge the MCO

Lines and valves to isolate the MCO include the isolation valves (and filters on air supply to valve actuators) in the VPS, general-service helium system, PWC system, and SCHe system. Upon demand, all the valves close to isolate the MCO, except the SCHe system valves, which open to allow helium to the MCO.

Safety-significant equipment for dilution and filtration to support SCHe operation

Process bay local exhaust HVAC and process vent system (exhaust fans and plenums, duct work, HEPA filters)

The process bay local exhaust HVAC and process vent system mitigates a gaseous release into the process bay by sweeping it through HEPA filters before it is discharged outside the facility. This ensures that SCHe flows are filtered and diluted.
Process bay local exhaust HVAC and process vent system process hood isolation damper and instrument air supply

Isolation dampers in the process bay local exhaust HVAC and process vent system process hood fail closed. If power is lost, dampers will open with electrical power from the standby power system and instrument air supplied by the local dedicated tank. The hood isolation damper and instrument air supply operate in conjunction with the standby power system to facilitate HVAC operating while on standby power. This supports operation of the local exhaust to filter and dilute SCHe flows.

Standby electrical power (diesel generator and process bay local exhaust HVAC and process vent system restart circuit)

The standby power system provides connections to restart the local exhaust fans and supporting equipment. Operation of the local exhaust on standby power will maintain building differential pressure sufficient for confinement during facility power outages. This supports operation of the local exhaust to filter and dilute SCHe flows.

Assumptions made that require protection by TSRs are listed below:

- Restore bay temperatures following process bay high temperature trip

  On high bay temperature trip alarm, operations must return the process bay temperatures to within acceptable limits.

T 3 Thermal runaway reaction in MCO due to loss of MCO control caused by facility fire

The accident scenarios identified in the hazard analysis for the T 3 bin represent thermal runaway accidents caused by process upsets from internal and external facility fires (HNF-SD-SNF-HIE-004). The controls in the T 3 bin prevent a thermal runaway by protecting safety-related SSCs from damage during a fire event in the bay. Credit also is taken for controls on the bay temperature to protect an assumption made in the fire hazards analysis.

Safety-class equipment for detection

- SCIC process bay high temperature trip

The SCIC process bay high temperature trip acts to protect an initial condition assumption in the fire hazards analysis.
Assumptions made that require protection by TSRs are listed below

- Combustible loadings limited

  While an MCO is present in the facility, combustible loadings are limited as determined by the fire hazard analysis implementation plan (SNF-4942)

- Restore bay temperature following a process bay high temperature trip

  On high bay temperature trip alarm, operations must return the process bay temperature to within acceptable limits

T4  Thermal runaway reaction in MCO due to loss of support utilities

  The accident scenarios identified in the hazard analysis for the T4 bin represent thermal runaway accidents caused by a loss of a support system because of accidents in adjacent bays, crane load drops, and accidents in the spare bay (HNF-SD-SNF-HIE-004) The controls in the T4 bin prevent a thermal runaway by monitoring key process parameters, precluding water addition to the MCO, and placing the MCO into a safe and stable configuration SCHe actuation is credited, therefore confinement and dilution provided by the local exhaust system also are credited

  Safety-class equipment to prevent process upsets

  - Deionized water and PWC drain line isolation valves interlocked closed after being used in the process

    The VPS isolation valves (on the deionized water supply line) and the PWC drain line isolation valves prevent water from entering the MCO After these lines are used in the process, they are closed and interlocked to the SCIC system so they cannot be inadvertently opened later in the process

T5  Thermal runaway reaction in MCO due to contamination of helium supply

  The accident scenario identified in the hazard analysis for the T5 bin represents a thermal runaway accident caused by fuel reactions with contaminants in the helium supply (HNF-SD-SNF-HIE-004) The controls in the T5 bin prevent a thermal runaway by controlling the helium supply to the MCO
Assumptions made that require protection by TSRs are listed below

- Shipment paperwork verified for gas bottle content during receipt

  The manufacturer's paperwork and shipping papers shall be checked upon helium cylinder receipt at the CVDF, both the normal and safety-class supply, to verify that the cylinder's contents were sampled by the supplier and that the sample met the required purity specification of >99% helium

T 6 Thermal runaway reaction in MCO due to line break caused by a seismic event

The accident scenarios identified in the hazard analysis for the T 6 bin represent thermal runaway accidents caused by process upsets following a seismic event (HNF-SD-SNF-HIE-004). The controls in the T6 bin prevent a thermal runaway by removing reliance on nonqualified SSCs and placing the MCO into a safe and stable configuration.

Safety equipment for detection of seismic event

- SCIC seismic trip

  On detecting a seismic event, the SCIC system seismic trip isolates the MCO and actuates the SCHe system and de-energizes the tempered water (annulus) system to ensure the seismic event will not initiate thermal runaway.

Assumptions made that require protection by TSRs are listed below

- Trailer placement controlled such that movement from seismic events will not impact key shutdown systems

  Before connecting CVDF systems to the MCO, the cask, which is located upon the trailer, must be positioned such that the cask is no greater than an established safe distance from the ideal horizontal placement. This control protects the MCO and safety-class components during seismic events.

B3 4 2 6 Multi-Canister Overpack Overpressurization Overpressurization accidents occur if there is no pressure vent or relief on a sealed MCO. The internal pressure in the MCO would increase until the MCO confinement failed or until all the fuel or water was consumed by the fuel-water chemical reaction. Detailed calculations of the consequences of an MCO overpressurization reaction are provided in SNF-2770, Chapter 7.0. The SSCs designated to prevent or mitigate MCO overpressurization are designated safety class because unmitigated consequences exceed offsite release limits.

Version 1 3 2 of the HANSF computer code (SNF-3650) was used to simulate the overpressurization cases. HANSF calculates the fuel, gas, and MCO structure temperatures and
MCO pressure based on uranium-water and hydride-water reactions, decay heat, and heat transfer. The code also calculates the particulate or MAR for the thermal cases over time. Quality assurance for the HANSF code at the Hanford Site is described in SNF-5226.

Safety-class features that prevent an MCO overpressurization accident include the SCIC system and instrumentation to detect process upsets and activate the SCHe system; the SCHe system to provide the MCO purge and vent function, the 30 lb/in² gauge pressure vent path, the 150 lb/in² gauge rupture disk, and the safety-class portions of the tempered water system. These preventive features reduce the estimated overpressurization accident frequency into the extremely unlikely range. Since the unmitigated offsite dose is well below the release limit of 5 rem for an extremely unlikely event, no additional safety-class mitigation features are required. However, success of some of the credited prevention features (e.g., the 30 lb/in² gauge rupture disk) might result in dose consequences that exceed onsite guidelines. Thus, safety-significant mitigation features provided for the MCO overpressurization accident include components of the process bay local exhaust HVAC system, process general exhaust HVAC system, associated HEPA filtration, process bay differential pressure alarms, and electrical standby power. The safety-significant mitigated dose for this accident is well below the applicable guideline.

**B3 4 2 6 1 Scenario Development** Based on the CVDF hazard analysis report (HNF-SD-SNF-HIE-004), several potential accidents have been identified that would lead to MCO overpressurization. From all of the potential overpressurization reaction accidents, a bounding accident was chosen for the unmitigated consequence analysis. The most credible high-pressure scenario consists of an isolated MCO with stationary annulus water. Without the safety features identified in this section, a loss of power would close the isolation valves and shut off the tempered water pump leaving the MCO isolated and without flowing annulus water.

The overpressurization DBA is assumed to begin after the water draining process is finished with some undrained, residual water still present in the MCO to react with the fuel. As little as 5 kg of residual water can pressurize the MCO vessel to 300 lb/in² gauge. This is caused by the residual water reacting with uranium and hydride, which produces hydrogen gas that pressurizes the isolated MCO. The greatest consequences are expected for the overpressurization accident with the highest pressure conditions. High internal gas pressure is required to cause a breach of MCO confinement and a blowdown release.

With uranium-water reaction rates producing hydrogen gas at all MCO temperatures, the overpressurization accident scenario will eventually lead to an increase in pressure sufficient to breach the MCO. An isolated MCO will reach unacceptable pressures given enough time (more than a week for cool tempered water annulus cases). The key aspect of high-pressure overpressurization accidents is the amount of time required to breach the MCO.

The MCO overpressurization scenario consists of the following events and processes:

- MCO pressurization until the isolated MCO is breached.
Blowdown release of particulate through an opening assumed to be equivalent to 1-in-diameter circle

Continuous release of gases and particulate from uranium–water reactions (the release continues as long as unreacted uranium is in contact with water vapor or oxygen)

The following assumptions are used in the analyses:

- The MCO overpressurization reaction scenario is assumed to begin when the water draining process is finished but some undrained (residual) water is still present in the MCO to react with the fuel. It is assumed that 26.5 kg of free undrained (residual) water (1.5 kg in the top scrap basket, 6 kg in the bottom scrap basket, 18 kg in three fuel baskets, and 1 kg in the gap between the bottom fuel basket and the MCO bottom plate) is present when the high-pressure overpressurization accident is initiated (HNF-SD-SNF-CN-023 [1998]). As little as 5 kg of residual water can pressurize the MCO vessel to 300 lb/in² gauge.

- Uranium hydrate contains 119 kg of water (HNF-SD-SNF-TI-015)

- The MCO is at bounding configuration with two scrap baskets:
  - Enhancement factor or rate multiplier fuel-water reaction equals 10 (HNF-SD-SNF-TI-015)
  - Surface area equals 12 m² (HNF-SD-SNF-TI-015)
  - Decay heat rate equals 776 W for five fuel baskets and 705 W for three fuel baskets and two scrap baskets (HNF-SD-SNF-TI-015)

- Uranium hydrides are included using an enhancement factor or rate multiplier of 12 to account for the mass fraction of hydride and its surface area (HNF-SD-SNF-TI-015).

- No hydrogen gettering takes place because of the presence of water during the entire scenario. Hydrogen gettering or uranium-hydrogen reaction ($U + 1.5\ H_2 \rightarrow UH_3$) has been shown not to occur to any significant degree in the presence of water or oxygen (SNF-5226).

- About 15 kg of particulate is generated between the time the MCO leaves the K Basins area and the time it is drained at the CVDF (HNF-SD-SNF-TI-015). This newly generated particulate is available to be entrained by exiting gases during a blowdown release and the subsequent long continuous release.
Annex B — Cold Vacuum Drying Facility

- Unmitigated blowdown release pressure is assumed to be 345 lb/in² gauge, which is greater than the MCO design pressure of 150 lb/in² gauge

- The bay temperature is assumed to be 90 °F

For the unmitigated MCO overpressurization accident sequence, the MCO vents or port valves are postulated to be closed or blocked. Thus, the uranium—water chemical reaction generates hydrogen gas, which pressurizes the MCO. The steam or water vapor from the heated liquid water also increases the MCO pressure (for an isolated MCO) and, more importantly, provides an oxidant for the exposed fuel in the scrap and fuel baskets. The increasing gas temperature in the MCO also helps pressurize the MCO. The blowdown at a pressure of 345 lb/in² gauge is assumed to occur about 96 hours after MCO isolation. This pressure would cause the rupture disk (rated at 150 lb/in² gauge) to burst and could also fail the MCO. However, to be conservative, the MCO is assumed to retain confinement until the 345 lb/in² gauge pressure is reached.

The MCO Topical Report (HNF-SD-SNF-SARR-005) states that based on the codes used in the MCO design, the mechanically closed MCO (before welding at the CSB) would not suffer any damage at internal pressures up to 340 lb/in² gauge. At some pressure in excess of 340 lb/in² gauge, the thread root of the collar would begin to yield, and at some pressure very much larger than 340 lb/in² gauge, leakage around the HelicoFlex seal would result. An MCO was built and tested to an internal pressure of 562.5 lb/in² gauge to match the test requirement to be placed on MCO procurements. In that test, no leakage and no permanent damage were observed. Credit is not taken for the pressure test, however, because the data is in draft form and the ramifications on selected controls have not been fully evaluated. Therefore, the MCO is assumed to be susceptible to failure at any pressure greater than 340 lb/in² gauge.

If the MCO were to be breached from high pressure, most of the gas and some suspended particulate would be carried out of the MCO in a very short time (<10 seconds). In addition to the particulate, hydrogen also would be released during the blowdown. The consequences of hydrogen releases are discussed in Section B3 4 2 3. The features and controls that prevent or mitigate the particulate release also prevent or mitigate the hydrogen consequence.

After the MCO is breached, the fuel-water reactions continue without the high pressure. Even with no air ingress into the MCO, the remaining water in the MCO keeps reacting with the exposed fuel, causing a continuous stream of gases and entrained particulate to flow out of the MCO. Both the scrap fuel and exposed fuel elements will react with the water vapor that is flowing upwards through the MCO and exiting out the orifice. The water vapor continues to react with uranium and uranium hydride until the water is completely consumed, which is more than 24 hours after the blowdown.

**B3 4 2 6 2 Source Term Analysis** The respirable source term is based on the MAR, the ARF and the RF. These terms are summarized in the following sections.
Material at Risk  In summary, the total particulate MAR consists of two distinct groups of particulate (UO₂) based on the two different types of releases (blowdown and continuous)

- MAR₁ = 60 kg UO₂ for the blowdown release, which is all of the particulate generated between the time the fuel is cleaned at the K Basins and the time of the blowdown. This equates to 52.9 kg of uranium.

- MAR₂ = 64 kg UO₂ for the 12-hour continuous release, which is all of the particulate newly generated after blowdown plus the new particulate created before blowdown and not released during blowdown (this latter amount is approximately equal to MAR₁, as only a small percentage is released during initial blowdown). This equates to less than 57 kg of uranium.

- MAR₃ = 69 kg UO₂ for the 24-hour continuous release, which is all of the particulate newly generated after blowdown plus the new particulate created before blowdown and not released during blowdown (this latter amount is approximately equal to MAR₁). This equates to about 61 kg of uranium.

Airborne Release Fraction and Respirable Fraction  The ARFs are different for each type of release. For the "venting of pressurized powders" with pressures less than 25 lb/in² gauge, the bounding respirable release fraction is 2 x 10⁻³, as reported by DOE-HDBK-3010-94 (Section 4.4.2.3.2, page 4-73). The respirable release fraction is multiplied by the MAR for the blowdown MAR₁ in order to obtain a particulate source term for the quantity of material released to the environment by the blowdown. The lower pressure values are used rather than the values listed for pressures greater than 25 lb/in² gauge, in order to avoid being overly conservative as discussed below.

The very large release fractions associated with high-pressure venting (pressures greater than 25 lb/in² gauge) are overly conservative for the MCO blowdown release and are not used. They are overly conservative because the high-velocity (initially sonic) gases in the MCO blowdown occur in the rupture disk orifice and shield plug and not underneath the particulate in the MCO as is the case in the high-pressure vented gas experiments described in DOE-HDBK-3010-94. Furthermore, the gas velocities inside the MCO baskets are low. At the time of blowdown, the gas velocities (averaged velocities inside the baskets, not at the basket base plate) range from 0.01 m/s near the MCO bottom, to 0.3 m/s at the lower part of bottom fuel basket (or upper part of bottom scrap basket), to 1 m/s at lower part of top scrap basket. The mass flow rates were derived using the HANSF code (SNF-3650). Flow velocities were calculated using the upstream flow rates calculated by the code. The velocities are higher in the holes of the basket base plates, ranging from 1.5 m/s to 15 m/s, but the particulate is not located in these holes. The particulate is located above these holes where the gas velocities are lower because of the larger flow area in the basket. All of these MCO blowdown velocities in the baskets are very low compared with the actual velocity (1.024 m/s) of the mixed gas in the orifice during the blowdown and compared with the velocities encountered under the powder in the high-pressure gas experiments in DOE-HDBK-3010-94 (Section 4.4.2.3.2, page 4-73).
DOE-HDBK-3010-94 presents two choices for release fractions under these conditions, one for low pressure (<25 lb/in² gauge) and one for relatively high pressures (>25 lb/in² gauge). The release fraction for low pressure conditions was selected even though failure occurs at higher pressures. This is because the experiments that investigated the release fractions were performed under high air flow conditions, whereas the MCO particulate will be exposed to low gas velocities during a blowdown. Thus, the smaller respirable release fraction, $2 \times 10^{-3}$, was assumed in this analysis.

In addition, the leak flow path is very tortuous in the MCO, especially in the scrap basket, and some particulate is located out of the gas flow path. Both of these characteristics reduce the ARF for the blowdown release out of the MCO but are not included in the source term analysis because of their calculational difficulty, and they were not included in the release experiments (DOE-HDBK-3010-94, Section 4.4.2.3.2).

The source term, $M_1$, from the blowdown release is calculated by multiplying $MAR_1$ by the product of ARF and RF and converting the particulate to uranium metal as follows:

$$M_1 = (MAR_1) (respirable \ release \ fraction)$$

$$= (52.9 \ kg \ U) (1,000 \ g/kg) (2 \times 10^{-3})$$

$$< 106 \ g \ U$$

The source term, $M_2$, for the 12-hour continuous release is calculated by multiplying $MAR_2$, the particulate MAR for the 12-hour continuous release, by the respirable release fraction ($4.8 \times 10^{-5}$) for gaseous release under debris for 12 hours (DOE-HDBK-3010-94, Section 4.4.1.2) as follows:

$$M_2 = (MAR_2) (respirable \ release \ fraction)$$

$$= (57 \ kg \ U) (1,000 \ g/kg) (4.8 \times 10^{-5})$$

$$< 3 \ g \ U$$

The source term, $M_3$, for the 24-hour continuous release (offsite) is calculated by multiplying $MAR_3$, the particulate MAR for the 24-hour continuous release, by the respirable release fraction for 24 hours ($9.6 \times 10^{-5}$) as follows:

$$M_3 = (MAR_3) (respirable \ release \ fraction)$$

$$= (61 \ kg \ UO_2) (1,000 \ g/kg) (9.6 \times 10^{-5})$$

$$< 6 \ g \ U$$
In summary, the total bounding source term (unmitigated) for environmental doses consists of three distinct groups based on the duration of the release:

- \( M_1 = 106 \text{ g U} \) for the blowdown release
- \( M_2 = 3 \text{ g U} \) for the 12-hour continuous release
- \( M_3 = 6 \text{ g U} \) for the 24-hour continuous release

### B3 4 2 6 3 Consequence Analysis

The radiological dose is calculated using the following equation:

\[
D_E = M \times \frac{\chi}{Q'} \times BR \times UD
\]

where

- \( D_E \) = effective dose equivalent based on inhalation exposure only (rem)
- \( M \) = respirable quantity released into the air (grams)
  - \( M_1 \) for blowdown release (106 g U)
  - \( M_2 \) for 12-hour (onsite) continuous release (3 g U)
  - \( M_3 \) for 24-hour (offsite) continuous release (6 g U)
- \( \chi/Q' \) = air transport factor (s/m³)
- \( BR \) = average inhalation rate during the blowdown and 12-hour release (m³/s)
- \( UD \) = committed effective dose equivalent per unit gram inhaled

#### Unmitigated Consequences

The dose calculation equation is used to calculate the dose to the offsite receptor:

\[
D_{\text{offsite1}} = M_1 \times \frac{\chi}{Q'} \times BR \times UD
\]

\[
= (106 \text{ g U}) (4.48 \times 10^5 \text{ s/m}^3) (3.33 \times 10^{-4} \text{ m}^3/\text{s}) (4.38 \times 10^5 \text{ rem/g U})
\]

\[
= 0.69 \text{ rem (0.0069 Sv) for blowdown release}
\]

\[
D_{\text{offsite3}} = M_3 \times \frac{\chi}{Q'} \times BR \times UD
\]

\[
= (6 \text{ g U}) (6.50 \times 10^6 \text{ s/m}^3) (2.64 \times 10^{-4} \text{ m}^3/\text{s}) (4.38 \times 10^3 \text{ rem/g U})
\]

\[
= 0.0045 \text{ rem (4.5 \times 10^{-3} Sv) for 24-hour continuous release}
\]

Adding the doses for the blowdown and continuous releases yields a total offsite dose of about 0.7 rem (0.007 Sv) with almost all of the dose coming from the blowdown release.

The dose consequences at the remaining receptor sites are calculated in the same manner and are shown in Table B3-19 for the blowdown and continuous release. The frequency of the unmitigated event is greater than \( 10^{-2} \) per year (see SNF-2770, Appendix A)
<table>
<thead>
<tr>
<th>Receptor Location (Distance from Process)</th>
<th>Mitigated dose limit (rem)</th>
<th>Mitigated evaluation limit (rem)</th>
<th>Unmitigated dose limit (rem)</th>
<th>Unmitigated evaluation limit (rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>100 m E</td>
<td>1.00 (1.1)</td>
<td>1.00 (1.1)</td>
<td>1.100 (1.1)</td>
<td>2.70 (0.027)</td>
</tr>
<tr>
<td>0.50 (0.045)</td>
<td></td>
<td>&quot;</td>
<td>&quot;</td>
<td>&quot;</td>
</tr>
<tr>
<td>100 m E</td>
<td>1.00 (1.1)</td>
<td>1.00 (1.1)</td>
<td>1.100 (1.1)</td>
<td>1.0 (1.0 E 2)</td>
</tr>
<tr>
<td>0.50 (0.045)</td>
<td></td>
<td>&quot;</td>
<td>&quot;</td>
<td>&quot;</td>
</tr>
</tbody>
</table>

**Notes:**
- The dose limits are for the High-Pressure Overpressurization Release scenario.
- The values are in rem (rems).
- The table is for the Columbia River and Hanford Site boundary.

**Additional Information:**
- Accident frequencies are provided for each receptor location.
- The table is from the HNF-3533 document, REV 0, Annex B — Cold Vacuum Drying Facility.

**References:**
- The methodology for calculating radiological and safety class decisions for unmitigated and mitigated doses is detailed in the referenced documents.

**Table B3.19:** Dose Consequence Summary for High-Pressure Overpressurization Release.
Mitigated Consequences  With safety-class features credited for this event (not including the 150 lb/in² gauge rupture disk), the estimated frequency for MCO overpressurization (without relief) is reduced to $4 \times 10^{-7}$ per year (beyond extremely unlikely category). The key safety-class features that reduce the likelihood of the overpressurization event are (1) the SCHe system, which provides pressure vent and helps cool the fuel, reducing the uranium–water reaction rate, (2) the 30 lb/in² vent path, which vents the MCO pressure, and (3) the 150 lb/in² MCO rupture disk, which relieves the MCO pressure. Additional safety-significant mitigation is provided by the HVAC systems and associated HEPA filters that reduce the onsite dose consequences from pressure venting. The safety-significant features reduce the mitigated onsite doses to well within applicable guidelines. All of the safety-class features are described in Section B3.4.2.6.5 along with the safety-significant features that provide additional mitigation for doses to the onsite receptor.

The safe and stable state following process upsets is an isolated MCO with pressure relief, a helium cover gas inside, and annulus water temperature at or below 50 °C (not necessarily flowing). If the annulus water is not flowing, the process bay temperature must be kept below 115 °F to keep fuel temperatures stable (SNF-2770).

If the SCHe system were assumed to not be available or not functional, the 30 lb/in² gauge vent path would vent the isolated MCO. The doses from the 30 lb/in² gauge vent path release meet offsite release limits guidelines when the local exhaust system with filters is functional. If both the SCHe system and 30 lb/in² gauge vent path were not available or not functional, the 150 lb/in² gauge rupture disk would relieve the isolated MCO. If, as expected, the local exhaust system filters remain at least 99% efficient under a hydrogen deflagration as expected (SNF-2770), the resulting doses from the 150 lb/in² gauge rupture disk release meet onsite risk evaluation guidelines and offsite release limits.

The scenarios for the releases at the following three pressures are summarized as follows:

1. The SCHe system is functional with 50 °C (or stationary) annulus water, functional local exhaust system with filters, and the MCO is isolated except for the 10 lb/in² gauge SCHe system. This is the normal configuration upon detection of a process upset. The frequency is in the anticipated category ($> 10^{-2}$ per year).

2. The SCHe system fails in a mode that isolates the MCO. The 30 lb/in² gauge vent path is functional with stationary annulus water, a bay temperature of 90 °F, functional local exhaust system with filters, and the MCO is isolated except for the 30 lb/in² gauge vent path. The frequency of this release sequence is estimated at $1.31 \times 10^{-4}$ per year, which is in the unlikely category.

3. Both the SCHe system and the 30 lb/in² gauge vent fail. Relief is provided through the 150 lb/in² gauge rupture disk. There is stationary annulus water, the bay temperature is 90 °F, the local and general exhaust systems with filters are functional, and the MCO is isolated except for 150 lb/in² gauge rupture disk.
The estimated frequency of this release sequence is $3.95 \times 10^{-7}$ per year (SNF-2770), which is beyond extremely unlikely.

The source terms for these scenarios are given in Table B3-20. The radiological doses are calculated in SNF-2770, and the total doses for several receptor locations with guidelines are shown in Table B3-21.

The SCHe system and normal operations are expected to prevent the MCO gas pressure from reaching 30 lb/in² gauge, thus, the 30 lb/in² vent path will not likely be used and is provided as backup to the SCHe system. The following section describes a scenario with two safety devices, the 30 lb/in² vent path, which includes the 30 lb/in² rupture disk and 30 lb/in² check valve, and stationary water in the cask-MCO annulus. Even if the water in the cask-MCO annulus is stationary, the thermal conductivity and heat capacity of the water are enough to keep the MCO at stable temperatures for process bay temperatures below 46 °C (115 °F) (SNF-2770). This is because (1) the stationary water conducts heat from the MCO to the cask better than air does (as in the loss of annulus water cases), and (2) the stationary water becomes cooler than the normal 50 °C annulus water at early times because of the cooling effects of the process bay temperature if the process bay temperature is less than 46 °C (115 °F).

The 30 lb/in² vent path with annulus water mitigated scenario assumes the process bay temperature is at 90 °F (32.2 °C). This scenario is the same as the unmitigated scenario except that the 30 lb/in² vent path is functional. The 30 lb/in² check valve is modeled to open at 35 lb/in² gauge pressure and close again at 20 lb/in² gauge pressure. The orifice size of the check valve was taken to be 0.2 in in the model, which is smaller than the actual real size of 0.25 in.

<table>
<thead>
<tr>
<th>Release duration</th>
<th>SCHe scenario</th>
<th>30 lb/in² gauge vent path scenario</th>
<th>150 lb/in² gauge rupture disk scenario</th>
</tr>
</thead>
<tbody>
<tr>
<td>Blowdown (&lt; 10 s)</td>
<td>NA</td>
<td>0.008</td>
<td>1.54</td>
</tr>
<tr>
<td>12 hours</td>
<td>0.0011</td>
<td>0.03</td>
<td>0.02</td>
</tr>
<tr>
<td>24 hours</td>
<td>0.0022</td>
<td>0.084</td>
<td>0.03</td>
</tr>
</tbody>
</table>

Note: Source terms are from SNF 2770 1999 Cold Vacuum Drying Facility Design Basis Accident Analysis Documentation Rev 3 Fluor Daniel Hanford Incorporated Richland Washington.
Table B3-21 Total Dose Summary from Mitigated Overpressurization Scenarios at Several Receptor Locations

<table>
<thead>
<tr>
<th>Receptor location (distance direction)</th>
<th>Evaluation guideline\textsuperscript{a} / release limit extremely unlikely\textsuperscript{b} rem (Sv)</th>
<th>SCHe system scenario total dose\textsuperscript{c} total dose\textsuperscript{d} rem (Sv)</th>
<th>30 lb/in\textsuperscript{2} gauge vent path scenario\textsuperscript{e} total dose rem (Sv)</th>
<th>150 lb/in\textsuperscript{2} gauge rupture disk scenario total dose rem (Sv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Onsite (100 m E)</td>
<td>25 (0.25)</td>
<td>1.0 E-03 (1.0 E-05)</td>
<td>0.12 (1.2 E-03)</td>
<td>16.4 (0.164)</td>
</tr>
<tr>
<td>Columbia River (650 m W)</td>
<td>3.2 E-05 (3.2 E-07)</td>
<td>0.0037 (3.7 E-05)</td>
<td>0.55 (0.0055)</td>
<td></td>
</tr>
<tr>
<td>100 Area Fire Station (3 750 m ESE)</td>
<td>4.2 E-06 (4.2 E-08)</td>
<td>3.2 E-04 (3.2 E-06)</td>
<td>0.036 (0.0036)</td>
<td></td>
</tr>
</tbody>
</table>

\textsuperscript{a} Evaluation guideline for onsite (100 m) receptor only
\textsuperscript{b} Unlikely event frequency is \( >10^4 \) to \( \leq 10^{-2} \)
\textsuperscript{c} Includes a decontamination factor of 1 E-03 for the SCHe system and 30 lb/in\textsuperscript{2} gauge consequences and 1 E-02 for the 150 lb/in\textsuperscript{2} gauge consequences to account for safety significant HEPA filters. Without credit for the HEPA filtration, the offsite consequences are below offsite release limits but the onsite doses for the 30 lb/in\textsuperscript{2} gauge release are above onsite risk evaluation guidelines.
\textsuperscript{d} Fifty year committed effective dose equivalent
\textsuperscript{e} The likelihood of a 150 lb/in\textsuperscript{2} gauge release is beyond extremely unlikely so comparison with guidelines and limits is not applicable. Consequences are provided for information only.

HEPA = high-efficiency particulate air (filter)
SCHe = safety-class helium

The blowdown from 150 lb/in\textsuperscript{2} gauge expels about 360 g of hydrogen gas in the first 10 seconds. The 12-hour continuous release will expel only an additional 27 g of hydrogen gas after the 10-second blowdown release, for a total of 387 g of hydrogen gas. At a pressure of 1 atm and a temperature of 267 °C, 360 g of hydrogen gas will occupy about 4 m\textsuperscript{3}. Diluting this volume to 20% hydrogen (a concentration that is high enough to produce destructive shock waves when it burns) requires a volume of air that is four times greater. Thus, the 360-g release needs to mix with about 16 m\textsuperscript{3} of air. For comparison, the CVDF process bays each have a volume of about 1,600 m\textsuperscript{3}, with enough fresh air added (>1 m\textsuperscript{3}/s from the local and general exhaust systems) to give more than two complete air changes per hour. Hence, flammable mixtures are possible, depending on the circumstances.

Actual events following the discharge of a large volume of hydrogen are difficult to predict. For the mitigated case, the local exhaust system is functioning and the likely events following the blowdown release are summarized below.

Most of the expelled hydrogen is drawn into the process bay local exhaust system. For the first 10 seconds, the hydrogen flow rate is about 0.4 m\textsuperscript{3}/s. This is about 80% of the nominal process bay local exhaust system flow rate (0.5 m\textsuperscript{3}/s [1,000 ft\textsuperscript{3}/min]). The hydrogen...
concentration will be very high in the duct. However, for the first 2 seconds of release, the hydrogen flow rate is much higher, so the nominal local exhaust rate will not capture all of the hydrogen. Some quantity of hydrogen will remain in the process bay after the blowdown starts. The sudden discharge of hydrogen into the local exhaust system will fill the process bay local exhaust system duct. When the hydrogen flow joins the air flow from the other bays, the hydrogen will be diluted to form an explosive mixture. An explosion of the hydrogen-air mixture near the HEPA filters would dislodge radioactive material and cause a fraction of it to be discharged to the environment. This accident is similar to the one of the external hydrogen explosions analyzed in the external hydrogen DBA (Section B3 4 2 3). Even if one bank of HEPA filters is completely destroyed, the second bank of HEPA filters is expected to be at least 99% efficient (SNF-2770).

With very good mixing, the hydrogen that did not get drawn in by the local exhaust system during the first 2 seconds becomes a flammable volume of 10 m³. If 50% of the hydrogen burns, the average pressure increase in the process bay is less than 0.35 lb/in² or 50 lb/ft². No significant structural damage is expected with this pressure. Using the method of estimating injury probabilities described in SNF-2770, death by lung hemorrhage has greater than a 1% chance, at a distance of 1.8 m (6 ft) from the center of the free-air explosion. A worker would need to be standing within this distance from the MCO head for this explosion to be fatal.

The continuous release of hydrogen for 12 hours after the blowdown release is slow enough (0.007 L/s on the average) that the hydrogen concentrations are easily diluted below the lower flammability limit by the process bay local exhaust system.

B3 4 2 6 4 Comparison to Guidelines

Comparison of Unmitigated Doses. Event tree sequences indicate that the unmitigated frequency for the MCO overpressurization accident is in the anticipated category (i.e., an unmitigated frequency greater than 10⁻⁴ per year) (see SNF-2770, Appendix A). The unmitigated sequence considered the processing of 200 MCOs per year, common cause loss of tempered water (annulus) system flow and an isolated MCO (loss of power). The unmitigated radiological offsite dose for this event is above offsite release limits and onsite risk evaluation guidelines.

Comparison of Mitigated Doses. The mitigated frequency of the event tree sequence that represents this DBA as an MCO overpressurization and as an unfiltered release is 4 × 10⁻⁵ per year, thus this mitigated DBA is extremely unlikely (see SNF-2770, Appendix A). The mitigated sequence credited SCHe injection, venting through the SCHe system vent path, relieving through the 30 lb/in² vent path, venting through the 150 lb/in² vent path, local exhaust operation and standby power, and HEPA filter functionality. With safety-significant features credited, the mitigated MCO overpressurization and unfiltered release meets both offsite release limits and onsite risk evaluation guidelines.
B3 4 2 6 5 Summary of Safety-Class Structures, Systems, and Components and Technical Safety Requirement Controls

Under normal operating conditions, annulus water levels and helium flow provide adequate heat transfer to prevent MCO overpressurization. Under upset or accident conditions, safety-class equipment is required in order to ensure this capability is not lost. The heat transfer characteristics of the annulus water could be compromised by a loss of water level or high tempered water temperatures. The heat transfer characteristics of the normal helium supply could be compromised by a loss of helium flow, high MCO pressures (e.g., isolated MCO), or low process system pressures (e.g., leaks).

The checklist designators included in the accident bins, other than the accident selected as the DBA, represent additional accident sequences slightly different than the DBA. All of these binned accidents are bounded by the DBA because they have lesser or equivalent worst-case consequences and frequencies.

The accident scenarios identified in the hazard analysis for the P1 bin represent overpressurization accidents caused by elevated fuel corrosion rates (high-temperature tempered water, excessive water remaining in an MCO because of proof-of-dryness demonstration failure [instrumentation failure or operator error]), inadvertent water addition to the MCO following final pressure rebound test, and loss of power because of flooding (HNF-SD-SNF-HIE-004). The controls in the P1 bin prevent an MCO overpressurization by monitoring the MCO pressure and by providing multiple vent paths from the MCO. Credit also is taken for controls that prevent bypass of safety systems by incorrectly proceeding with the cold vacuum drying process or by operator error. To protect the analysis assumption pertaining to the maximum rate of hydrogen generation, credit is taken for SSCs to control fuel temperatures. Credit also is taken for controls that prevent malfunction of safety-related SSCs (high bay temperature). In addition, SCHе actuation is credited, therefore confinement and dilution provided by the local exhaust system also are credited. The filtration function is also credited with mitigating a blowdown of the 30 lb/in² gauge vent path. Note that the 150 lb/in² gauge MCO rupture disk is identified in the analysis and shown in Table B3-22, but that the accident would be adequately controlled (prevented to beyond extremely unlikely) without crediting this rupture disk.

The safety-class equipment designated to prevent or mitigate the dose consequences of overpressurization accidents are described below.

Safety-class equipment for detection of process upset

- VPS instrumentation (high pressure trip)

Pressure indicators on the VPS provide MCO internal pressure information to the SCIC system, which initiates MCO isolation and SCHе actuation during process upset conditions.
<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator*</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4 0)</th>
<th>NRC ITSb</th>
</tr>
</thead>
</table>
| P 1 Overpressurization due to internal process upset of key parameters (bounding accident, PB B 13c) | PB B-03a | Prevent MCO overpressurization | Safety class equipment for detection and prevention of process upset  
VPS instrumentation (high pressure trip)  
Tempered water (annulus) system temperature trip  
Tempered water (annulus) system level alarm  
Tempered water (annulus) level check petcocks  
High bay temperature trip  
SCIC system  
Desonized water and PWC drain line isolation valves interlocked closed after being used in the process | B |
| | PB B 13c | Maintain parameters within limits | Safety class equipment for heat removal  
Tempered water (annulus) system piping and antusphon valves | B |
| | PB B 13d | | Safety-class equipment for pressure management  
SCHe system (vent path)  
30 lb/in² rupture disk and vent path lines and valves  
150 lb/in² rupture disk | B |
| | PB H 11f | | Safety-class equipment for confinement purge and pressurization  
Cask–MCO  
SCHe system  
Lines and valves to isolate and purge the MCO | A B B |
| | OU R 02 | | Safety significant equipment for confinement and dilution  
HVAC/PV system (exhaust fans and plenums duct work, HEPA filters and flow switch)  
HVAC/PV process hood isolation damper  
Standby electrical power (diesel generator and HVAC/PV system restart circuit)  
HVACD system (exhaust HEPA filter exhaust duct work, isolation damper)  
HVACB isolation dampers (outside air inlets) | B |
| | OU R 04 | | Safety significant equipment for monitoring  
Reference air system (reference air header differential pressure alarm for process bays) | |
| | | | TSR  
Procedure to verify the results of the pressure rebound tests before continuing process steps  
Close long axial process tube port before closing filtered process exit port at the end of processing | |
<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator*</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4 0)</th>
<th>NRC ITS³</th>
</tr>
</thead>
<tbody>
<tr>
<td>P 1 Overpressurization due to internal process upset of key parameters (cont)</td>
<td>PB F 02a SB F-01b SB F 02b OU P 04 OU R 03</td>
<td>Prevent MCO overpressurization Place the MCO in a safe configuration during a loss of support utilities</td>
<td>Defense in-depth equipment for detection General service helium system safety class flow instrumentation 10 lb/in² gauge safety relief valve for process relief</td>
<td>B</td>
</tr>
<tr>
<td>P 2 Overpressurization due to loss of support utilities</td>
<td></td>
<td></td>
<td>Safety-class equipment for pressure management SCHe system (vent path) 30 lb/in² gauge rupture disk and vent path lines and valves 150 lb/in² gauge rupture disk Safety-class equipment for confinement and purge Cask-MCO SCHe system Lines and valves to isolate and purge the MCO Safety significant equipment for confinement and dilution HVAC/PV system (exhaust fans and plenums duct work, HEPA filters and flow switch) HVAC/PV process hood isolation damper Standby electrical power (diesel generator and HVAC/PV system restart circuit) HVACD system (exhaust HEPA filter exhaust duct work, isolation damper) HVACB isolation dampers (outside air inlets) Safety significant equipment for monitoring Reference air system (reference air header differential pressure alarm for process bays)</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Defense in-depth equipment 10 lb/in² gauge safety relief valve for process relief</td>
<td></td>
</tr>
<tr>
<td>Candidate accident</td>
<td>Checklist designator*</td>
<td>Safety function</td>
<td>Safety features (described in Chapter B40)</td>
<td>NRC ITSb</td>
</tr>
<tr>
<td>---------------------</td>
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</tr>
<tr>
<td>P 3 Overpressurization due to excessive helium supply pressure</td>
<td>PB H 11a</td>
<td>Prevent excessive helium pressure inside the MCO</td>
<td>Safety-class equipment for pressure boundary Cask-MCO</td>
<td>A</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Safety class equipment for pressure control General service helium supply system safety-class relief valves SCHHe supply pressure control valves and rupture disks</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Defense in depth Pressure relief valve on tube transporter SCHHe vent 30 lb/in² rupture disk, vent path line and valves 150 lb/in² rupture disk</td>
<td></td>
</tr>
<tr>
<td>P 4 Overpressurization due to line break caused by a seismic event</td>
<td>PB R 01a OU R 01a</td>
<td>Prevent MCO overpressurization Place the MCO in a safe configuration following a seismic event</td>
<td>Safety class equipment for pressure management SCHHe system (vent path) 30 lb/in² gauge rupture disk and vent path lines and valves 150 lb/in² gauge rupture disk</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Safety class for detection of seismic event SCIC seismic trip</td>
<td>B</td>
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<td></td>
<td></td>
<td></td>
<td>Safety class equipment for confinement, purge and pressurization Cask-MCO SCHHe system Lines and valves to isolate and purge the MCO*</td>
<td>A B B</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Safety class equipment for heat removal Tempered water (annulus) system piping and antisiphon valves</td>
<td>B</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>TSR Trailer placement controlled such that expected movement from seismic events will not impact key shutdown systems</td>
<td></td>
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<td></td>
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<td></td>
<td>Defense in depth 10 lb/in² gauge safety relief valve for process relief</td>
<td>B</td>
</tr>
</tbody>
</table>

Annex B — Cold Vacuum Drying Facility

November 1999
<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designtator*</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4 0)</th>
<th>NRC ITS$^a$</th>
</tr>
</thead>
<tbody>
<tr>
<td>P 5 Overpressurization caused by facility fire</td>
<td>PB L 01</td>
<td>Protect a processing process bay against external fire (administrative area, transfer corridor other nonprocessing process bay)</td>
<td>Safety class equipment for detection SCIC process bay high temperature trip</td>
<td>B</td>
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<tr>
<td></td>
<td>PB L 02</td>
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<td>PB P 02</td>
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<td>OU P 02a</td>
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<tr>
<td></td>
<td></td>
<td>limit the fire risk inside a processing process bay</td>
<td>TSR</td>
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<td></td>
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<td></td>
<td>Combustible loadings limited</td>
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<td></td>
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<td>Restore bay temperatures following process bay high temperature trip</td>
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<td>Defense in depth equipment for process upset conditions and process bay temperature detection</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>VPS instrumentation (pressure) SCIC vacuum cycle timer (8 4-4 logic)</td>
<td></td>
</tr>
<tr>
<td></td>
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<td></td>
<td>Tempered water (annulus) system temperature trip Tempered water (annulus) system level alarm</td>
<td></td>
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<td></td>
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<td>General service helium system safety-class flow instrumentation</td>
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<td></td>
<td></td>
<td>Defense in depth equipment for heat removal Tempered water (annulus) system piping and antisiphon valves Manual refill piping and vent port Cask-MCO</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Defense in depth equipment for shutdown SCHe system Lines and valves to isolate and purge the MCO$^a$</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>Defense in depth Procedure to limit combustible loading Fire protection system present in each bay SCHe system Lines and valves to isolate and purge the MCO$^a$</td>
<td></td>
</tr>
</tbody>
</table>
### Table B3-22  Summary of Safety Features Required to Mitigate or Prevent an Overpressurization Reaction  (5 sheets)

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator</th>
<th>Safety function</th>
<th>Safety features (described in Chapter B4.0)</th>
<th>NRC ITS&lt;sup&gt;a&lt;/sup&gt;</th>
</tr>
</thead>
</table>

Checklist designators are from HNF SD SNF HIE-004 1999 *Cold Vacuum Drying Facility Hazard Analysis* Rev 4 Fluor Daniel Hanford, Incorporated, Richland Washington

<sup>a</sup>U.S. Nuclear Regulatory Commission important to safety classifications Category A = critical to safe operation, Category B = major impact on safety Category C = minor impact to safety

<sup>b</sup>Lines and valves to isolate the MCO include the isolation valves (and filters on air supply to valve actuators) and the air filters on the isolations valves in the VPS general service helium system, PWC system and SCHe system

- HEPA = high efficiency particulate air (filter)
- HVAC = heating, ventilation, and air conditioning
- HVACC/PV = process bay local exhaust HVAC and process vent system
- HVACB = process bay recirculation HVAC system
- HVACD = process general supply/exhaust HVAC system
- ITS = important to safety
- MCO = multi canister overpack
- NRC = U.S. Nuclear Regulatory Commission
- SCHe = safety-class helium
- SCIC = safety class instrumentation and control
- TSR = technical safety requirement
- VPS = vacuum purge system
Annex B — Cold Vacuum Drying Facility

- SCIC system (monitoring process parameters)

  The SCIC system uses programmable logic controllers, wiring to process instrumentation, signals from seismic detectors and temperature monitors, system controls, and output relays to isolate the MCO and actuate the SCHe system or cut power to the tempered water heater, if abnormal process or environmental conditions are detected.

- Tempered water (annulus) system temperature trip

  The tempered water (annulus) system includes antisiphon valves, low-water-level indication and alarm, and manual refill capability to ensure a minimum water level is maintained above the elevation of the fuel within the MCO. The system detects high water temperature and signals the SCIC system to actuate the tempered water trip.

- Tempered water (annulus) system low level alarm

  Redundant liquid level indicators are used by the tempered water (annulus) system to detect a low water level in the cask-MCO annulus and to provide a signal to the SCIC system to actuate the low-level alarm.

- Tempered water (annulus) system level check petcocks

  The level check petcocks provide a manual means of verifying the presence of water in the annulus. The petcocks are located in and above the double-walled portion of the tempered water (annulus) piping and are connected to the outer and inner pipes above the height of the fuel in the MCO. If a low-level alarm is received, the petcocks can be used to verify the presence of water in the inner or outer pipe, thus indicating the presence of water in the annulus.

- High bay temperature trip

  The SCIC high bay temperature trip isolates the MCO and actuates the SCHe system so excessive temperatures in the bay cannot cause instrument inaccuracies or malfunctions that might result in MCO overpressurization.

Safety-class equipment for heat removal

- Tempered water (annulus) system piping and antisiphon valves

  A portion of the tempered water (annulus) system contains safety-class piping and antisiphon valves to ensure retention of a minimum water level above the elevation of the SNF inside the MCO.
Safety-class equipment for pressure management

- **SCHe system (vent path)**
  
  SCHe system piping and valves provide a pressure vent path to the local exhaust system precluding high pressures in the MCO

- **VPS 30 lb/in² gauge rupture disk and vent path lines and valves**
  
  The VPS 30 lb/in² gauge rupture disk and valves in the vent path to the local exhaust duct preclude high pressures

- **MCO 150 lb/in² gauge rupture disk**
  
  The MCO 150 lb/in² gauge rupture disk on the MCO provides backup capability for the 30 lb/in² gauge rupture disk on the VPS vent path line. The two rupture disks are connected or installed in different MCO process ports. Releases through the 150 lb/in² gauge rupture disk would be discharged to the process bay and swept through HEPA filters before being released to the environment.

Safety-class equipment for confinement and purge

- **Cask–MCO**
  
  The cask–MCO is a major part of the pressure boundary for confinement of radioactive materials during processing and provides connections for the process piping to the SCHe system

- **SCHe system**
  
  The SCHe system provides two redundant and independent paths for purging and pressurizing the MCO and venting to the process vent

- **Lines and valves to isolate and purge the MCO**
  
  Lines and valves for isolating the MCO include the isolation valves (and air filters on air supply to valve actuators) in the VPS, general-service helium system, PWC system, and SCHe system. Upon demand, all the valves close to isolate the MCO, except the SCHe system valves, which open to allow helium to the MCO

Safety-significant equipment for confinement and dilution

- **Process bay local exhaust HVAC and process vent system (exhaust fans and plenums, ductwork, HEPA filters, and flow switch)**
The process bay local exhaust HVAC and process vent system mitigates a release by sweeping it through HEPA filters before it is discharged outside the facility. Low flow alarms are set at 1,150 ft³/min.

- Process bay local exhaust HVAC and process vent system process hood isolation damper and instrument air supply

Isolation dampers in the process bay local exhaust HVAC and process vent system process hood fail closed. If power is lost, dampers will open with electrical power from the standby power system and instrument air supplied by the local dedicated tank. The hood isolation damper and instrument air supply operate in conjunction with the standby power system to facilitate HVAC operating while on standby power.

- Reference air system (reference air header, differential pressure alarms)

The reference air system monitors the negative pressure in the process bays and process water tank room by providing differential pressure indication and alarms to the control room for operator response.

- Standby electrical power (diesel generator and process bay local exhaust HVAC and process vent system restart circuit)

The standby power system provides connections to restart the local exhaust fans and supporting equipment. Operation of the local exhaust on standby power will maintain building differential pressure sufficient for confinement during facility power outages.

- Process general supply/exhaust HVAC system (exhaust HEPA filter, exhaust ductwork isolation damper)

The process general supply/exhaust HVAC system mitigates a release into the process bay or process water tank room by sweeping it through HEPA filters before discharging it outside the facility. The process general supply/exhaust HVAC system also provides confinement in conjunction with the facility structure by maintaining a negative building pressure. The system's isolation dampers fail closed to maintain negative differential pressure in the processing bays and process water tank room.

- Process bay recirculation HVAC system isolation dampers (outside air inlets)

The process bay recirculation HVAC system provides fail-closed outside air inlet dampers so the local exhaust on standby power can maintain process bay differential pressure.
Assumptions made that require protection by TSRs are listed below

- **Procedure to verify the results of the pressure rebound tests before continuing process steps**

  An initial pressure rebound test surveillance (pressure rise test) must be met before entry into the proof-of-dryness demonstration is allowed. Similarly, a proof-of-dryness demonstration surveillance must be met before the final pressure rebound test steps can begin. Finally, a final pressure rebound test must be met before shipment preparation steps can begin.

- **Close long axial process tube port before closing filtered process exit port at the end of processing**

  When isolating the MCO at the end of processing, the long axial process tube port must be closed at least 5 minutes before closing the filtered process exit port. This ensures that an SCIC trip will occur if the procedure to isolate the MCO is inadvertently conducted on an MCO still to be processed.

**Defense in depth features**

- **General-service helium system safety-class flow instrumentation**

  The general-service helium system provides SCHe flow information to the SCIC system to initiate MCO isolation and SCHe actuation during process upset conditions.

- **10 lb/in² gauge safety relief valve for process relief**

  A general-service safety relief valve set at 10 lb/in² gauge and located on the process equipment skid provides pressure relief to the local exhaust duct.

  The bounding MCO overpressurization accident (P 1) and the other accidents identified in the CVDF hazard analysis report (HNF-SD-SNF-HIE-004) that can potentially involve MCO overpressurization are itemized in Table B3-22, along with corresponding checklist designators, safety functions, and SSCs.

  The accidents within the remaining bins require the following safety SSCs and TSRs in addition to the ones identified for the DBA (P 1).

**P 2 Overpressurization due to loss of support utilities**

The accident scenarios identified in the hazard analysis for the P 2 bin represent overpressurization accidents caused by loss of a support system because of accidents in adjacent
bays and accidents in the spare bay. This bin also includes loss-of-power events caused by external forces (e.g., vehicle accident) and lightning strike (HNF-SD-SNF-HIE-004). The controls in the P2 bin prevent an MCO overpressurization by providing multiple vent paths from the MCO. Monitoring of process parameters is not needed for the events in this bin because the SSCs that provide venting of the MCO fail safe on loss of support utilities. Confinement and dilution are credited because of SCHe actuation.

No additional requirements result from analysis of this accident.

P 3 Overpressurization due to excessive helium supply pressure

The accident scenario identified in the hazard analysis for the P 3 bin represents an overpressurization accident caused by failure of the pressure regulators on the helium supply (general service helium or SCHe) (HNF-SD-SNF-HIE-004). The controls in the P3 bin prevent an MCO overpressurization by controlling the helium supply pressure from the general service helium system and the SCHe system.

Safety-class equipment for pressure control

- General-service helium supply system safety-class relief valves

  Redundant pressure relief valves are provided on the general-service helium supply line to protect the MCO from overpressurization. General-service helium is supplied from the tube transporters at between 2,600 and 3,200 lb/in² gauge. The two pressure relief valves, which are arranged in series and set at 25 lb/in² gauge, prevent the MCO and related piping from experiencing the full supply pressure in the event of regulator failure.

- SCHe supply pressure control valves and rupture disks

  The SCHe system provides two redundant and independent paths for purging and pressurizing the MCO. Pressure control valves and rupture disks in each train protect the MCO from overpressurization by the helium bottles.

P 4 Overpressurization due to a line break caused by a seismic event

The accident scenarios identified in the hazard analysis for the P 4 bin represent overpressurization accidents caused by process upsets following a seismic event (HNF-SD-SNF-HIE-004). The controls in the P4 bin prevent an MCO overpressurization by removing reliance on nonqualified SSCs and by placing the MCO into a safe and stable configuration.
Safety-class equipment to prevent process upsets

- Deionized water and PWC drain line isolation valves interlocked closed after being used in the process

The VPS isolation valves (on the deionized water supply line) and the PWC drain line isolation valves prevent water from entering the MCO. After these lines are used in the process, they are closed and interlocked to the SCIC system so they cannot be inadvertently opened later in the process.

Safety-class equipment for detection of seismic events

- SCIC seismic trip

On detecting a seismic event, the SCIC system seismic trip isolates the MCO and actuates the SCHe system and de-energizes the tempered water (annulus) system to ensure the seismic event will not initiate thermal runaway.

Assumptions made that require protection by TSRs are listed below

- Trailer placement controlled such that movement from seismic events will not impact key shutdown systems

Before connecting CVDF systems to the MCO, the cask, which is located upon the trailer, must be positioned such that the central axis of the cask is no greater than an established safe distance from the ideal horizontal placement, as identified in Chapter B4. This control protects the MCO during seismic events.

P 5 Overpressurization due to facility fire

The accident scenarios identified in the hazard analysis for the P 5 bin represent overpressurization accidents caused by facility fire (HNF-SD-SNF-HIE-004). The controls in the P5 bin prevent an MCO overpressurization by protecting safety-related SSCs from damage in a process bay fire. Credit also is taken for controls on the bay temperature to protect an assumption made in the fire hazards analysis.

Safety-class equipment for detection

- SCIC process bay high temperature trip

The SCIC process bay high temperature trip acts to protect an initial condition assumption in the fire hazards analysis.
Assumptions made that require protection by TSRs are listed below

- **Combustible loadings limited**

  While an MCO is present in the facility, combustible loadings are limited as determined by the fire hazard analysis implementation plan (SNF-4942). These limits ensure that any fire in the CVDF does not result in uncontrolled releases (e.g., fire-caused loss of process control).

- **Restore bay temperature following a process bay high temperature trip**

  On high bay temperature trip alarm, operations must return the process bay temperature to within acceptable limits.

**B3 4 2 7 Common Cause Initiators** As part of the hazard analysis process, several common cause initiators were identified that have the potential to impact the CVDF (HNF-SD-SNF-HIE-004). These initiators have been addressed as discrete accidents within the respective accident bins but are discussed collectively below to address the issue of common cause failure risks at the CVDF.

**B3 4 2 7 1 Seismic Forces** Earthquake forces could affect systems across the entire facility. Without controls and safety SSCs, a seismic event could cause process upsets ranging from a liquid release in the PWC tank room to thermal runaway reactions within MCOs in the processing bays. It is unlikely that at any one point in time all MCOs in the facility will be susceptible to a thermal runaway reaction DBA or the overpressurization DBA, but to be conservative, the consequences of four MCOs experiencing such an event can be postulated in the unmitigated (i.e., no controls) case. In addition, other effects may be possible (e.g., gaseous release, external hydrogen explosion, PWC spray leak).

The bounding integrated facility doses for the unmitigated seismic event are approximately equal to the total doses from four design basis overpressurization accidents (the overpressurization DBA has the greatest unmitigated consequences). In the unmitigated case, the dose consequences of an external explosion and spray release from the PWC room are negligible compared to the overpressurization events. From the dose consequence summary table, Table B3-6, the unmitigated onsite consequences of four overpressurization events would be approximately 4,400 rem, and the offsite consequences would be approximately 28 rem. These doses clearly exceed both onsite and offsite criteria. The overpressurization accident analysis identified safety-class, performance category 3 equipment as a preventative feature. This equipment is presented in the seismic bins for the overpressurization DBA. In addition, safety-class, performance category 3 (performance category is used in the design of SSCs for resistance to natural phenomena) equipment is identified for the thermal runaway DBA, and safety-class, performance category 3 and safety-significant, performance category 2 equipment is identified for the internal hydrogen explosion (this event has safety-significant consequences but some equipment is safety-class, performance category 3 because it is credited in another DBA).
For the external hydrogen explosion DBA, a TSR limitation on HEPA filter loading is credited in the seismic event bin. From Table B3-6, with the credited controls applied, the mitigated dose consequences of a seismic event are reduced to approximately 10 rem onsite and $6 \times 10^4$ rem offsite. These doses correspond to the sum of the mitigated doses associated with four gaseous releases, an external hydrogen explosion, and a spray leak in the PWC room. All of the other DBAs are prevented to beyond extremely unlikely and have no associated dose consequences for the mitigated case. The mitigated dose consequences for the seismic event fall within offsite release limits and onsite risk evaluation guidelines. Therefore, the credited controls specified in the DBA analyses (seismic bins) are sufficient to prevent a release following a seismic event.

**B3 4 2 7 2 Loss of Facility Power** Natural phenomena events, such as rain, lightning, and freezing weather, can result in the temporary loss of facility power. In addition, certain external events such as vehicle impacts with components outside the facility could also result in a loss of facility power. Without controls, a loss of power can result in a loss of process control from the control room and subsequent upset conditions. Such upsets could potentially lead to several DBAs, including the thermal runaway reaction. However, loss of power does not result in any type of liquid spray or spill within the facility. For example, a loss of power could result in isolation of an MCO and subsequent overpressurization or failure to repressurize an MCO that is under vacuum.

The integrated facility doses for this event are bounded by those presented in the seismic forces discussion in Section B3 4 2 7 1. The instrumentation credited with detecting process upsets is designed to fail safe upon a loss of power, activating the necessary systems to bring the MCO to a safe and stable state (SChE, isolation valves and lines, SCIC power cutoff to the tempered water [annulus] heater). In addition, the facility has a standby power system designed to provide electrical power to key facility SSCs. As demonstrated, the bounding mitigated consequences are below all dose consequence criteria, so the controls specified in the DBA analyses for each bay are adequate to protect against a loss of facility power.

**B3 4 2 7 3 High Process Bay Temperatures** High process bay temperatures could affect certain systems within the process bays. The safety-related instrumentation associated with the tempered water (annulus) system is designed to survive high temperatures, but some of the instrumentation related to MCO process upset detection and activation of the SChE system could under abnormal or accident conditions, potentially experience temperatures above rated design limits. Without controls, this could result in instrumentation inaccuracies and lead to internal or external MCO explosions or place the MCO into unanalyzed conditions. For example, a high bay temperature could lead to elevated annulus water temperature and subsequent loss of heat sink for the MCO. High temperatures do not pose a hazard for the PWC tank room.

The integrated facility doses for the unmitigated event are bounded by those presented in Section B3 4 2 7 1. Safety-class equipment (process bay temperature detection instrumentation SChE system, and the lines and valves necessary to isolate the MCO) is identified in the DBA analysis to prevent process upsets caused by high bay temperatures. If high bay temperatures are...
reached, this equipment will place the MCOs in a safe and stable condition. A safety-class high-bay-temperature trip alerts facility personnel to the bay condition so bay temperatures can be restored. No release occurs in this mitigated case. Therefore, the controls identified in the DBA analyses for each bay are sufficient to prevent a release caused by high bay temperatures.

**B3 4 2 7 4 Loss of Facility Support Systems** A loss of facility support systems such as the normal helium supply, instrument air, and HVAC flow could impact multiple bays and systems within the CVDF. Support system loss can result in process upsets that can lead to unfavorable conditions in the MCO (e.g., a low helium flow, contaminated helium supply, and inaccurate instrumentation readings). A loss of ventilation flow does not result in any dose consequences, however, it does reduce the margin of safety should other events occur while the system is inoperable. A loss of facility support systems also is not expected to cause any type of spill or spray release in the PWC tank room. In the process bays, if multiple MCOs were being processed, the common cause effect would be multiple upset conditions.

The integrated facility doses for this event are bounded by those presented in Section B3 4 2 7 1. The equipment and controls for protection against process upsets will place the facility in a safe and stable state when process upsets occur. As demonstrated, the bounding mitigated consequences are below all dose consequence criteria, so that the controls specified in the DBA analyses for each bay are adequate to protect against a loss of support systems.

**B3 4 2 7 5 Multiple Bay Facility Fire** A facility fire has the potential to have impacts in multiple bays and to serve as an initiator for more than one of the DBAs in Section B3 4 2. Individual fire scenarios were developed and analyzed in the CVDF fire hazard analysis (SNF-4268) and associated implementation plan (SNF-4942), and these fires are considered as accident initiators in the applicable DBAs. Fires, as accident initiators, are controlled through the fire protection features designed into the facility and operating equipment, and through the combustible loading limitations (supported by the SCIC high temperature trip) as presented in the fire hazard analysis. In the DBA analysis, credit is taken only for the combustible loading limits presented in the fire hazard analysis and the SCIC high temperature trip. These controls preclude the potential consequences from a single fire involving multiple bays. The integrated facility doses for a worst-case multiple bay facility fire are bounded by those presented in the seismic forces discussion in Section B3 4 2 7 1.

**B3 4 2 7 6 Emergency Evacuation Due to External Events** Should the CVDF need to be evacuated because of external events at other facilities, a loss of operator control over the process could occur. Although the facility can be brought to a safe and stable condition prior to evacuation through emergency response procedures in such an event, the inherent risk is assessed without such procedures. Lack of operator control represents another initiator for a loss of process control. Without controls, this event could result in process upsets leading to DBAs in one or more of the process bays.

The integrated facility doses for the unmitigated event are bounded by those presented in Section B3 4 2 7 1. The safety-class equipment for protection against process upsets.
automatically place the facility in a safe and stable state when process upsets occur by placing each bay into a safe and stable state. There is no operating mode that requires operator action to place the process in a safe and stable state. Operator actions are credited in the DBA analysis, but such actions afford time delays that accommodate evacuation actions. However, no events were identified in which operator actions were credited in conditions that would also require evacuation. Therefore, a facility evacuation is adequately protected against with the existing controls identified within the DBAs.

**B3 4 3 Beyond Design Basis Accidents**

DOE-STD-3009-94 requires consideration of BDBAs for nuclear facilities but presents no clear guidelines as to how or to what criteria they should be evaluated and judged. The Standard indicates that the purpose of BDBA presentation is to gain insight into the magnitude of these events, particularly if they should be close in frequency to pertinent DBAs but beyond the credible range and if they should have consequences that exceed evaluation guidelines. While these events would be beyond the requirements for further safety-class or safety-significant functions, they might provide guidance for prioritizing long-term safety improvements for a facility. The Standard specifically excludes evaluation of human-generated external events as BDBAs.

The BDBA analysis for the CVDF included the following:

- **Extreme cold** — Breach of an MCO caused by excessively cold weather and resulting ice formation
- **Natural phenomena** — Collapse of the CVDF building superstructure as a result of natural phenomena forces of a severity larger than those comprising the design basis of the facility
- **Unmitigated DBAs** — Potential accident conditions exceeding the capabilities of the identified safety systems
- **Shipment to the CSB of excess free water in an MCO**

Each of these BDBA types is described below.

The evaluation of the BDBAs identified for the CVDF indicates that there is no sharp increase in dose consequences from the design basis to the beyond design basis, and thus no sharp increase in residual risk. No additional controls to mitigate the residual BDBA risk were identified.

**B3 4 3 1 Extreme Cold** During extremely cold weather conditions, the water inside the MCO or in the cask–MCO annulus could freeze. This event could rupture the MCO shell resulting in loss of the pressure boundary. This could prevent processing of the MCO because of the inability...
to maintain annulus water (i.e., draining the MCO also drains the annulus) and the inability to draw a vacuum on the MCO. If the cask also were to rupture, a continuous release of contamination could occur following a loss of annulus and MCO water (i.e., they both drain once the ice melts). In the worst case, this would be equivalent to the thermal runaway event (Section 3.4.2.5), so no significant increase in potential dose consequences would be realized from the beyond design basis event.

**B3 4 3 2 Natural Phenomena** A natural phenomenon BDBA involves a natural phenomenon force greater than the design basis of the facility. The two dominant natural phenomena forces for the CVDF, seismic and high wind forces, would result in similar structural impacts to the facility. However, a high wind would also result in very large dispersion factors for atmospheric transport and significantly lower dose consequences. Therefore, the natural phenomenon BDBA postulated for the CVDF is a beyond design basis seismic event.

In accordance with the codes and standards used for the CVDF, the facility SSCs are designed with significant safety margins. Thus, complete failure of all SSCs is not expected during or following a seismic event of a magnitude slightly larger than the design basis. However, there is a possibility that a severe beyond design basis seismic event with a very low return frequency could impact the facility and cause numerous SSC failures, including building structural failure. In such an event, any cask-MCO packages present in the facility would be impacted by failed facility structural components, and systems connected to the MCO would be damaged. In the worst-case scenario, the MCO could be overturned, and all process equipment, including the process hood and process connections, could be dislodged from the MCO.

Even under severe impacts from facility structural components, the cask-MCO package would not fail catastrophically because of its very strong physical design. However, some damage to the MCO could be realized and both process ports could be open or closed, depending on the process step at the time of the event. The bounding consequences from such an event would be for all four bays to have MCOs that had been drained but still had sufficient free water remaining in them to present thermal runaway and overpressurization hazards. This case is addressed in the common mode accident evaluation for seismic events in Section B3 4.2.7.1. This potential beyond design basis consequence does not constitute a large increase over the design basis consequences.

**B3 4 3 3 Unmitigated Design Basis Accidents** The unmitigated scenario analyzed for each DBA in Section B3 4.2 represents a beyond design basis event. Each unmitigated scenario represents failure of the credited controls through an internal mechanistic failure (e.g., isolation valve fails to open on demand) or through an external condition (e.g., a physical impact). By representing events and conditions that exceed the design basis (i.e., conditions in which SSCs do not function properly), the analyses of unmitigated events represent beyond design basis events. As demonstrated in the DBA analyses, the unmitigated events are adequately controlled.

**B3 4 3 4 Excess Free Water Following Cold Vacuum Drying Facility Processing** One event of particular interest is the shipping of an MCO containing more than the allowed amount of free water.
water to the CSB. The consequences of such an event would occur as a result of a deficiency in
CVDF operations but logically would not be an accident occurring at the CVDF because the
accident (i.e., overpressurization of an MCO) would occur at the CSB, not at the CVDF. This
scenario is postulated to occur as a result from the failure of validation and acceptance testing of
an MCO after it has been subjected to the cold vacuum drying process or from a late, inadvertent
addition of water after completing the testing but prior to the MCO leaving the CVDF. The
unmitigated consequences of this event would be similar to the unmitigated consequences
presented in Section B3 4 2 6 for the MCO overpressurization DBA. The frequency of shipping
an MCO from the CVDF to the CSB with greater than 200 g due to a process failure or
inadvertent water addition has been analyzed (SNF-2770). The conclusion of the analysis is that
the frequency of this event is beyond extremely unlike (or less than $1 \times 10^{-6}$ per year) is based on
the (1) CVDF operational procedures that include disconnecting deionized water lines, drying,
and recognizing excessive drying time, (2) redundant isolation valves on potential water sources,
and (3) redundant temperature measurement of the tempered water temperature. The cold
vacuum drying process and validation test are designed so that retention of free water (in excess
of the allowable quantity) following processing is not a credible occurrence.

B3 5 REFERENCES

10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and

ALCOA, 1987, Oxides and Hydroxides of Aluminum, ALCOA Technical Paper No. 19, Revised,
ALCOA Laboratories, Pittsburgh, Pennsylvania

Vacuum Drying Facility, Contract No. MRV-SBW-482901, Task 15, Science Applications
International Corporation, Richland, Washington

DOE/EIS-0245F, Addendum (Final Environmental Impact Statement) Management of Spent
Nuclear Fuel from the K Basins at the Hanford Site, Richland Washington


DOE Order 5480 22, Technical Safety Requirements, U.S. Department of Energy,
Washington, D.C.

DOE Order 5480 23, Nuclear Safety Analysis Reports, U.S. Department of Energy,
Washington, D.C.


Annex B — Cold Vacuum Drying Facility


November 1999


SNF-5226, 1999 *Comparison Cases Simulated with HANSF 1 3 2 to Supplement Thermal Analyses Documented in HNF-SD-SNF-CN-023*, Rev 0, Fluor Daniel Hanford, Incorporated, Richland, Washington


Figure B3-1 Hazards/Accident Analysis Flow Diagram
(Adaptation of Figure 3-1 from DOE-STD-3009-94)

Type form location and quantity,

3311 Hazard Identification

3312 Hazard Classification

Determine appropriate method of hazard evaluation (PHA, HAZOP, etc.)

Graded approach input

3312 Hazard Evaluation

Identify by type (fire, explosion, etc.) and by category (operational, natural phenomena, and external)

Basic accidents

33231 Planned design and operational safety improvements

33232 Defense in depth

33233 Facility worker safety

33234 Environmental Protection

Is facility Hazard Category 3?

Yes

accident analysis not necessary

Identification of safety-significant SSCs and TSRs

No

33235 Accident Selection

Exception would be a chemical release hazard approaching Evaluation Guidelines

Design Basis Accident Analysis

3 4 2 X 1 (Typical) Scenario Development

3 4 2 X 2 (Typical) Source Term Analysis

3 4 2 X 3 (Typical) Consequence Analysis

3 4 2 X 4 (Typical) Comparison to EGs

3 4 2 X 5 (Typical) Identification of SSC/SSC9S and TSRs

Design Basis Accidents Selected for CVDF

3 4 1 Methodology

3 4 2 Design Basis Accidents

3 4 2 1 Gaseous Release

3 4 2 2 Liquid Release

3 4 2 3 MCO Internal Hydrogen Explosion

3 4 2 4 MCO External Hydrogen Explosion

3 4 2 5 MCO Thermal Runaway Reaction

3 4 2 6 MCO Overpressurization

3 4 2 7 Common Cause Initiators

3 4 3 Beyond Design Basis Accidents

November 1999
### Figure B3-2  Hazardous Material/Energy Source Checklist  Example

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18 **Signed**

L J Besser

Signature of EDI Originator 1/28/99

19 **Authorized**

L J Besser

Authorized Representative Date for Receiving Organization 1/28/99

20 **Cognizant Manager**

R J Morgan

Design Authority/ Cognizant Manager 1/28/99

21 **DOE APPROVAL**

[] Approved

[] Approved w/comments

[] Disapproved w/comments
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Spent Nuclear Fuel Project, Final Safety Analysis Report

L J Garvin
Fluor Daniel Northwest Inc, Richland, WA 99352
U S Department of Energy Contract DE-AC06 96RL13200

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Abstract The Spent Nuclear Fuel (SNF) Project Final Safety Analysis Report (FSAR) is a multi volume document Volume 1 contains SNF Project information applicable to the Canister Storage Building Cold Vacuum Drying Facility and 200East Interim Storage Area that are generic to the SNF Project This volume contains 17 chapters as required by DOE STD 3009 Preparation Guide for U S Department of Energy Nonreactor Nuclear Facility Safety Analysis Report

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Volume 1

SPENT NUCLEAR FUEL PROJECT
FINAL SAFETY ANALYSIS REPORT

HNF-3553
Revision 0

November 1999
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EXECUTIVE SUMMARY

E 1 FACILITY BACKGROUND AND MISSION

The U.S. Department of Energy (DOE) established the Spent Nuclear Fuel (SNF) Project to address safety and environmental concerns associated with deteriorating SNF presently stored under water in the Hanford Site K Basins, which are located in the 100 K Area near the Columbia River. Recommendations for a series of projects to construct and operate systems and facilities to manage the safe removal and storage of K Basins fuel were made in WHC-EP-0830, *Hanford Spent Nuclear Fuel Recommended Path Forward* and its subsequent update WHC-SD-SNF-SP-005, *Hanford Spent Nuclear Fuel Project Integrated Process Strategy for K Basins Fuel*. The integrated process strategy recommendations include the following steps:

- Fuel preparation activities at the K Basins, including removing the fuel elements from their K Basins canisters, separating fuel particulate from fuel elements and fuel fragments greater than 0.25 in. in any dimension, removing excess sludge from the fuel fragments by means of flushing, as necessary, and packaging the fuel into multi-canister overpacks (MCOs).

- Transportation of MCOs loaded with SNF from K Basins to the Cold Vacuum Drying Facility (CVDF).

- Removal of free water by draining and vacuum drying at the CVDF in the 100 K Area.

- Dry shipment of fuel from the CVDF to the Canister Storage Building (CSB), a new facility in the 200 East Area.

- Interim storage of the MCOs in the CSB until a suitable long-term repository is established.
E 2 SPENT NUCLEAR FUEL PROJECT OVERVIEW

WHC-SD-SNF-SP-005 was developed to establish the technical framework for constructing facilities and implementing processes that are compatible with DOE goals. This process strategy subsequently evolved into the current SNF Project technical baseline. The current technical baseline is based on implementation of Baseline Change Request SNF-98-006, which describes the process and design approaches for removing, drying, and storing the K Basins SNF (Sieracki 1998).

SNF Project process steps include removing K Basins SNF assemblies from existing containers, cleaning the fuel, placing the fuel into specially designed baskets, and inserting the baskets into approximately 400 MCOs. The MCOs are filled with water and transported to the CVDF, where the water is drained and the MCOs are vacuum dried. MCOs are then sealed by mechanical means, leak tested, and shipped to the CSB where they are received, sampled, and permanently sealed by a welding process. The MCOs will be held in interim storage in vertical tubes at CSB until final disposition. A small number of the first MCOs shipped to the CSB will be monitored over a period of up to two years to validate that the process meets long-term storage requirements.

For more detailed discussions of the SNF Project facilities and processes, see Chapters 20 of the facility annexes to the SNF Project Final Safety Analysis Report (FSAR).

E 3 FACILITY HAZARD CLASSIFICATION

The final facility hazard classification of each SNF Project facility was performed in accordance with DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23 Nuclear Safety Analysis Reports*. The final hazard categorizations are based on unmitigated releases of available radiological materials from each
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facility These inventories were compared against threshold quantities contained in DOE-STD-1027-92 Each SNF Project facility was classified as Hazard Category 2

E 4 SAFETY ANALYSIS OVERVIEW

Hazard identification and evaluation provides a thorough assessment of the spectrum of risks posed by a nuclear facility to the public, workers and the environment. A hazard analysis determines materials and systems and handling, processing, storage, and facility characteristics that can produce undesirable consequences as a result of equipment malfunctions, operator errors, manmade hazards, and/or natural phenomena hazards. Predominantly qualitative techniques are used to pinpoint weaknesses in the design or operation of the facility that could lead to accidents. A selection process is used to identify design basis accidents with the potential to cause unacceptable consequences to the offsite and onsite receptor groups and to the environment. The design basis accidents are further analyzed using quantitative methods and a set of controls is selected to ensure that the facility can be operated safely. The accident analysis for each design basis accident starts with a description of the accident scenario and the identification of major analysis assumptions. The accident source term is then determined through phenomenological and system response calculations. Once the source term has been determined, onsite and offsite consequences are determined and the consequences are compared to offsite release limits and onsite evaluation guidelines. Finally, a set of safety-class and safety-significant equipment and technical safety requirements (TSRs) is selected to ensure safe facility operation and worker protection.

E 5 U S DEPARTMENT OF ENERGY REGULATORY POLICY

The regulatory policy for the SNF Project (Sellers 1995) delineates the DOE policy for achieving nuclear safety equivalence for the design and construction of new facilities comparable to facilities licensed by the U.S. Nuclear Regulatory Commission (NRC).
E 5 1 U S Department of Energy Policy for Safety of New Facilities

DOE facility nuclear safety equivalence to comparable NRC-licensed facilities is accomplished by applying comparable technical requirements and by adopting appropriate features of the NRC licensing process in addition to applicable DOE orders and requirements. DOE requirements are not set aside or superseded by this process, rather, along with applicable NRC regulations, the DOE requirements and NRC regulations comprise the basis for SNF Project facility design and construction.

The intended outcome of the Nuclear Safety Policy (Sellers 1995) is to ensure that SNF Project facility designs meet the nuclear safety objectives of applicable NRC requirements. From a nuclear safety standpoint, rugged conservative designs, with clear capability to accommodate postulated accidents, provide the best confidence that the SNF Project facility designs are equivalent to NRC-licensed facility designs.

The regulatory policy for the SNF Project is to be applied to matters of nuclear safety (including radiological control issues) for new SNF Project facilities. The primary focus is on design and construction issues and preparation for operations. The policy does not apply to environmental Occupational Safety and Health Administration chemical accident safety or other nonnuclear safety issues. (These issues are covered elsewhere by DOE orders and statutory requirements.) Similarly, the life-of-facility oversight (e.g., operator training, performance assessment) applied by the NRC to NRC-licensed facilities is not covered by the SNF Project regulatory policy.

The following definitions are used in applying the SNF Project regulatory policy to new SNF Project facilities.
Safety equivalence to NRC-licensed facilities

For the purpose of the SNF Project regulatory policy, safety equivalence to NRC-licensed facilities is established for SNF Project facilities by conformance to the following technical and administrative requirements:

- Technical requirements that meet the nuclear safety objectives of NRC regulations for fuel treatment and storage facilities, including requirements regarding radiation exposure limits, safety analysis, design, and construction.

- Administrative requirements that meet the objectives of the major elements of the NRC licensing process, including formally documented design and safety analyses, independent technical reviews, and opportunities for public involvement.

NRC requirements for comparable facilities

The term "requirements" means design and construction measures that are specifically mandated by NRC regulations. Regulatory guidance and precedents, which are illustrative of implementation of the regulations, are considered optional rather than mandatory. The term "comparable facilities" in this case means SNF treatment and storage facilities.

E 5.2 Requirement Development Process

The proposed SNF Project regulatory requirements were identified by the contractor with the help of the Regulatory Requirements Team. The requirements development process included high-level screening of NRC regulations (10 CFR Parts 0 through 199) to select those applicable to the SNF Project facilities, comparison with DOE requirements to identify significant areas of...
difference and compilation of the composite set of requirements to be applied to the SNF Project. NRC regulatory guidance was used as needed to clarify the intent of the NRC regulations and to provide insight into suitable methods of implementation. The results of the Regulatory Requirements Team’s review were presented in WHC-SD-SNF-DB-002, *Spent Nuclear Fuel Project Path Forward Nuclear Safety Equivalency To Comparable Nuclear Regulatory Commission Licensed Facilities*.

Because the fuel form (metal) used at the Hanford Site differs from the normal U.S. commercial reactor fuel (metal oxide), engineering analysis was used where necessary to develop new (or modified) requirements in areas where neither the NRC or DOE requirements explicitly addressed SNF Project technical issues.

Each NRC technical safety equivalence requirement proposed for adoption was reviewed against the criteria provided in the DOE Nuclear Safety Policy (Sellers 1995). This review was conducted to ascertain whether the addition of a requirement would result in a cost-effective increase in safety.

In some cases the requirements were presented in the form of white papers that provided the arguments and discussions that supported the position of nuclear safety equivalency. For example, the white paper for seismic design criteria is documented in WHC-SD-SNF-DB-004, *Spent Nuclear Fuel Project Seismic Design Criteria NRC Equivalency Evaluation Report*.

The additional requirements as developed were agreed upon by the Regulatory Requirements Team, which was operating as a consensus technical body. The DOE Richland Operations Office approved the requirements with the concurrence of the Independent Review Panel and the DOE Office of Environmental Management, Office of Environment and Safety and Health. These requirements are documented in HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward Additional NRC Requirements*.
E 5 3 Final Safety Analysis Report Documentation of the Requirements

The SNF Project facility NRC equivalency process described above was initiated early in the life of the SNF Project before the facility designs had progressed beyond the conceptual stage. The additional NRC requirements identified as designs progressed were transmitted to the facility design engineering firms for implementation. Because some requirements were identified during the early state of the designs, some items were included that are not applicable to the final designs. As the designs evolved, the SNF Project subprojects developed and then updated matrices identifying how the various NRC requirements were applicable to the current designs. The various chapters of the subproject FSARs included as annexes to the SNF Project FSAR, contain references to the applicable matrices and summarize how each facility meets the applicable NRC requirements.

The CSB FSAR, Annex A, includes additional content that has been extracted from NRC Regulatory Guide 3 48, Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage). The CVDF FSAR, Annex B includes additional content that has been extracted from Regulatory Guide 3 26 Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants. This additional content is required by item 24 of HNF-SD-SNF-DB-003. The additional content requirement is further documented in HNF-SD-SNF-SP-012. Additional Guidance for Including Nuclear Safety Equivalency in the Canister Storage Building and Cold Vacuum Drying Facility Final Safety Analysis Reports. HNF-SD-SNF-SP-012 contains tables that delineate the additional NRC content on a chapter-by-chapter basis using the guidance provided in DOE-STD-3009-94 Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports, for organizing the safety analysis reports. The applicable chapters of the CSB and CVDF FSARs (Annexes A and B) contain this additional information in the appropriate DOE-STD-3009-94 location.
E 6 ORGANIZATIONS

Fluor Daniel Hanford, Incorporated (FDH) is responsible to the DOE for planning, integrating, and managing SNF Project activities including programs, projects, and operations. FDH is supported by subcontractors, including Duke Engineering and Services Hanford Incorporated. As management contractor, FDH is ultimately responsible for contract performance, which includes protecting the public workers, and the environment from hazards associated with Hanford Site operations. FDH also acts as a focal point for interaction with DOE and stakeholders and provides an Environment, Safety, and Health (ES&H) oversight organization. The ES&H organization is responsible for developing and implementing sitewide programs and for providing support to the SNF Project in areas such as occupational safety and health, emergency preparedness, nuclear safety, environmental protection, and radiation protection. The FDH Vice President in charge of ES&H reports to the FDH President. An FDH Vice President and Project Director is responsible for leadership, oversight, and control of the SNF Project and also reports to the FDH President. Other FDH sitewide entities with significant supporting relationships to the SNF Project are the Quality Assurance and Training Organizations.

The SNF Project has an established organizational structure, defined responsibilities, assigned interfaces, policies, and programs that support safe design, construction, and operational activities of SNF Project facilities. The SNF Project supports the mission to clean up the Hanford Site by managing and reducing hazards associated with the Hanford Site SNF inventory.

The SNF Project has responsibility for the design, construction, and operation of the SNF Project facilities. The SNF Project organization includes the Design Authority (Engineering), Operations, and Maintenance functions and has primary responsibility for executing the project mission. This includes defining systems, managing subprojects and programs, providing technical direction to design agents and construction managers, reviewing and approving products and activities, and ensuring that requirements are met. As the modifications to K Basins and construction of the CVDF and CSB are completed, the SNF Project will transition from its...
present role as an organization involved primarily with design, construction, surveillance, and maintenance to a processing role that will involve the activities of removing the SNF from K Basins, transporting the SNF to CVDF, conditioning the SNF at CVDF, and transporting the SNF to the CSB.

The SNF Project Engineering Manager and Chief Engineer has overall responsibility for establishing, implementing and maintaining the engineering technical baseline, facility authorization basis (including the FSAR and TSRs) and nuclear safety regulatory compliance for SNF Project facilities. As Design Authority for the SNF Project, the Chief Engineer defines requirements for and ensures the technical adequacy of all facility structures, systems and components. The Chief Engineer is also responsible for ensuring that the initial releases of all design and safety basis documentation are technically sound and that changes to these documents are consistent with the approved authorization basis. Within the SNF Project Engineering organization, the Nuclear Safety group is tasked with providing overall coordination and integration of SNF Project regulatory matters, preparing all authorization basis documents for the SNF Project facilities, and managing the unreviewed safety question evaluation process for SNF Project facilities. Other significant functions within the Engineering organization include Process Engineering (responsible for development of technical baseline documents and validation and monitoring of fuel storage processes) and Facilities Engineering (responsible for system upgrades and modifications).

The SNF Project Operations organization is responsible for managing and directing SNF Project operational activities in a safe and environmentally sound manner. This organization is charged with conducting all of their activities in compliance with DOE contractual orders and applicable federal, state and local laws. These activities include facility operations such as handling and storage of SNF, maintenance, start-up and testing, and operational support functions such as management of the occupational radiation safety program, training of facility operators and performance of operational readiness reviews.
Several internal and external organizations provide independent performance-based and compliance-based reviews of the SNF Project. These organizations are as follows:

- **Quality Assurance** – Independent assessments will be performed by the cognizant SNF Project Quality Assurance function to ensure that the requirements of the SNF Project quality assurance program plan are being satisfied. The types and frequencies of assessments performed will be based on the status, complexity, importance, and past performance of the activity or process being assessed.

- **Safety Review Board** – The primary responsibilities of the Safety Review Board are to provide a formal review and approval process for safety basis documents and to ensure adherence to the requirements of DOE orders and standards. Safety Review Board members are appointed by the SNF Project Director for a minimum term of one year. Safety basis documents to be reviewed include safety analysis reports, TSR documents, operational readiness reviews, fire hazard analyses, and criticality safety evaluation reports.

- **Facility Evaluation Board** – Members for the Facility Evaluation are provided by the FDH sitewide Quality Assurance organization. The Facility Evaluation Board is comprised of individuals not otherwise associated with the activity or process being assessed. The scope of a Facility Evaluation Board independent assessment includes radiological controls, occupational safety and health quality assurance environmental programs, engineering training configuration management, maintenance, nuclear safety, fire protection, and operations. The purpose of Facility Evaluation Board assessments is to establish facility compliance with external and internal requirements and to identify areas for improvement in the areas assessed.

- **Regulatory Requirements Team** – A Regulatory Requirements Team to meet requirements of the DOE Regulatory Policy comprises members from contractor.
organizations, DOE organizations, and outside consultants, all of whom are selected for their knowledge in areas relevant to the SNF Project. The Regulatory Requirements Team reports to the DOE Richland Operations Office, Spent Fuels Division Project Director. The purpose of the Regulatory Requirements Team is to identify, develop, and propose for approval selected regulatory requirements applicable to the SNF Project Facilities and to the fuel products removed from the K Basins.

- Independent Review Panel – An Independent Review Panel reporting to the Office of the Manager of DOE, Richland Operations Office, established to meet the requirements of the DOE Regulatory Policy, provides advice and high-level oversight of the implementation of regulatory policy with relation to the SNF Project. The Independent Review Panel comprises three members selected from outside the DOE complex based on their technical capabilities and experience applicable to the SNF Project. The Independent Review Panel is charged with review of SNF Project regulatory strategy, evaluation and concurrence with SNF Project regulatory requirements, and evaluation and concurrence with SNF Project safety analysis and safety evaluation reports, including those submitted for final approval to operate the SNF Project facilities.

E 7 SAFETY ANALYSIS CONCLUSIONS

For safety analyses conclusions for SNF Project facilities, consult Chapters 3 0 of the facility FSAR Annexes.
E 8 FINAL SAFETY ANALYSIS REPORT ORGANIZATION

The SNF Project FSAR is a multi-volume document. This Volume 1 contains SNF Project information applicable to all new project facilities. Volumes 2, 3, and 4 are facility FSAR Annexes A, B, and D, which contain the CSB CVDF, and 200 Area Interim Storage Area FSARs, respectively. The chapters in all four volumes of the FSAR follow the seventeen-chapter DOE-STD-3009-94 format. Chapters and sections in the SNF Project FSAR are numbered as 10111220, 30, and so forth. Chapter numbers in the facility FSAR Annexes contain the prefix of the respective annex. For example, the chapter and section numbers in Annex A, the CSB FSAR, are identified as A10, A11, A12, A20, A30, and so forth.

This report is based on the format and content guidance of DOE-STD-3009-94 and the requirements of DOE Order 5480 23 Nuclear Safety Analysis Reports. This report is also based on implementation of NRC nuclear safety equivalency as presented and approved in HNF-SD-SNF-DB-003.

On a chapter-by-chapter basis, the information contained in this Volume 1 of the SNF Project FSAR, is summarized as follows:

Chapter 10 contains descriptions of the Site Characteristics of the region in which all three SNF Project facilities are located, including geography, land and water uses, demography, meteorology, and natural phenomena hazards.

Chapter 20 includes Facility Description information that is common to all three SNF Project facilities. Most of this type of information is however facility-specific in nature and is found in the facility FSAR Annexes.

Chapter 30 describes requirements and methodology used to perform Hazard and Accident Analyses for the SNF Project facilities. Actual analyses are described in the facility FSAR Annexes.
Chapter 4 0  Describes requirements and methodology used to select Safety Structures Systems and Components for the SNF Project facilities Actual selected controls are described in the facility FSAR Annexes

Chapter 5 0  Describes requirements and methodology relevant to the Derivation of Technical Safety Requirements Actual elaboration of TSRs and their relationships to hazard and accident analyses and to safety equipment are described in the facility FSAR Annexes

Chapter 6 0  Describes requirements and methodology relevant to Prevention of Inadvertent Criticality, and contains characterizations of SNF inventories at SNF Project facilities Individual conclusions regarding criticality and related controls are found in the facility FSAR Annexes

NOTE  In those cases where policies programs and practices important to safe operation are described in detail in other documents the information is summarized in the programmatic chapters below and the documents are referenced The detailed programs and procedures described in referenced documents may be changed without further DOE approval to the extent that the changes do not constitute an unreviewed safety question as defined by DOE Order 5480.21 Unreviewed Safety Question

Chapter 7 0  Contains a description of the Radiation Protection program that applies to the SNF Project facilities

Chapter 8 0  Contains a description of the Hazardous Material Protection program that applies to the SNF Project facilities

Chapter 9 0  Contains a description of the Radioactive and Hazardous Waste Management program that applies to the SNF Project facilities
Chapter 10 0 Contains a description of the *Initial Testing  In-Services Surveillance and Maintenance* program that applies to the SNF Project facilities.

Chapter 11 0 Contains a description of the *Operational Safety* program that applies to the SNF Project facilities.

Chapter 12 0 Contains a description of the *Procedures and Training* program that applies to the SNF Project facilities.

Chapter 13 0 Describes requirements and methodology relating to evaluation and incorporation of *Human Factors* into the SNF Project facility designs. Specific human factors evaluations and design features are found in the facility FSAR Annexes.

Chapter 14 0 Contains a description of the *Quality Assurance* program that applies to the SNF Project facilities.

Chapter 15 0 Contains a description of the *Emergency Preparedness* program that applies to the SNF Project facilities.

Chapter 16 0 Describes some of the general types of design features and overall approach taken by the SNF Project facilities for *Provisions for Decontamination and Decommissioning*. Specific design features and other measures to ensure effective decontamination and decommissioning are described in the facility FSAR Annexes.

Chapter 17 0 Describes the *Management Organization and Institutional Safety Provisions* that characterize the SNF Project. Some of these organizational features specifically those that relate to aspects of safety management are also addressed in Section E 5 of this Executive Summary.
REFERENCES

10 CFR Parts 0 through 199, "Energy" Code of Federal Regulations

DOE Order 5480 21, Unreviewed Safety Questions, U S Department of Energy, Washington D C

DOE Order 5480 23, Nuclear Safety Analysis Reports, U S Department of Energy, Washington, D C


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<td>AC</td>
<td>Administrative Control</td>
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<tr>
<td>ACGIH</td>
<td>American Conference of Governmental Industrial Hygienists</td>
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<tr>
<td>ALARA</td>
<td>as low as reasonably achievable</td>
</tr>
<tr>
<td>AMS</td>
<td>Office of Assistant Manager for Engineering and Standards</td>
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<td>BED</td>
<td>building emergency director</td>
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<td>building emergency plan</td>
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<td>Canister Storage Building</td>
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HAZWOPER hazardous waste operation
HEHF Hanford Environmental Health Foundation
HFD Hanford Fire Department
HFE human factors engineering
HGET Hanford General Employee Training
HMI human-machine interface
HMS Hanford Meteorological Station
HR-NR Hog Ranch-Naneum Ridge
HWVP Hanford Waste Vitrification Plant
IAA integrated audit and appraisal
IC incident commander
ICP Incident Command Post
ID69 Ident-69
IRP Independent Review Panel
ISA Internm Storage Area
ISC interim storage cask
ISMP Integrated Safety Management Plan
ISMS integrated environment, safety, and health management system
IWTS integrated water treatment system
IXM ion exchange module
JHA job hazard analysis
JIC Joint Information Center
JSA job safety analysis
LCO Limiting Condition for Operation
LCS Limiting Control Setting
LWR light water reactor
M&I Management and Integration
M&TE measuring and test equipment
Ma million years ago
MCO multi-canister overpack
MSDS material safety data sheet
MTU metric ton of uranium
NAC Nuclear Assurance Corporation
NEPA National Environmental Policy Act (NEPA) of 1969
NFPA National Fire Protection Association
NOAA National Oceanic and Atmospheric Administration
NRC U S Nuclear Regulatory Commission
O&M operations and maintenance
OCRWM Office of Civilian Radioactive Waste Management
ONC Occurrence Notification Center
ORP U S Department of Energy, Office of River Protection
ORR operational readiness review
OSHA Occupational Safety and Health Administration
OWL Olympic-Wallowa lineament
PAG protective action guide
PAR protective action recommendation
PAT preoperational acceptance test
PEL permissible exposure limit
PHA preliminary hazard analysis
PHMC Project Hanford Management Contract
PHMS Project Hanford Management System
PMP probable maximum precipitation
POC Patrol Operations Center
PPE personal protective equipment
QA quality assurance
QAPP quality assurance program plan
QARD quality assurance requirements and description
RAMA reliability, availability, and maintainability analysis
RAW Rattlesnake Mountain to Wallula Gap
RCRA Resource Conservation and Recovery Act of 1976
RFP request for proposal
RL U.S. Department of Energy, Richland Operations Office
RRT Regulatory Requirements Team
RWP radiological work permit
S/RID standards/requirements identification document
SAR safety analysis report
SED Site Emergency Director
SFO U.S. Department of Energy Office of Spent Nuclear Fuels
SMT Site Management Team
SNF spent nuclear fuel
SPR single pass reactor
SRB Safety Review Board
SRS Savannah River Site
SSC structure, system and component
TEDE total effective dose equivalent
TLV threshold limit value
TPA Tri-Party Agreement
TRIGA Test Reactor and Isotope Production General Atomics
TSR technical safety requirement
UDAC Unified Dose Assessment Center
USQ unreviewed safety question
UW University of Washington
VPP Voluntary Protection Program
WAC Washington Administrative Code
WCC Woodward-Clyde Consultants
WMH Waste Management Federal Services of Hanford Incorporated
WNP Washington Nuclear Plant
YFB Yakima Fold Belt

November 1999
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CHAPTER 10
SITE CHARACTERISTICS
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<tr>
<td>CLEW</td>
<td>Cle Elum-Wallula</td>
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<tr>
<td>CRBG</td>
<td>Columbia River Basalt Group</td>
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<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
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<tr>
<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
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<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
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<tr>
<td>FSAR</td>
<td>Final Safety Analysis Report</td>
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<tr>
<td>HMS</td>
<td>Hanford Meteorological Station</td>
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<tr>
<td>HR-NR</td>
<td>Hog Ranch-Naneum Ridge</td>
</tr>
<tr>
<td>HWVP</td>
<td>Hanford Waste Vitrification Plant</td>
</tr>
<tr>
<td>Ma</td>
<td>Million years ago</td>
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<tr>
<td>NEPA</td>
<td>National Environmental Policy Act</td>
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<tr>
<td>NOAA</td>
<td>National Oceanic and Atmospheric Administration</td>
</tr>
<tr>
<td>NRC</td>
<td>U.S. Nuclear Regulatory Commission</td>
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<tr>
<td>OWL</td>
<td>Olympic-Wallowa Lineament</td>
</tr>
<tr>
<td>PMP</td>
<td>Probable Maximum Precipitation</td>
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<tr>
<td>RAW</td>
<td>Rattlesnake Mountain to Wallula Gap</td>
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<tr>
<td>SNF</td>
<td>Spent Nuclear Fuel</td>
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<tr>
<td>SRS</td>
<td>Savannah River Site</td>
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<td>SSC</td>
<td>Structure System and Component</td>
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<td>UW</td>
<td>University of Washington</td>
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10 SITE CHARACTERISTICS

11 INTRODUCTION

The K West and K East Basin Fuel Storage Facilities and the Cold Vacuum Drying Facility (CVDF) are located in the 100 K Area of the U.S. Department of Energy (DOE) Hanford Site. The CVDF is used to remove free water from spent nuclear fuel (SNF) taken from the K West and K East fuel storage basins and to vacuum dry the SNF before it is transferred to the Canister Storage Building (CSB) in the 200 East Area. The objective of this chapter is to describe the characteristics of the region, Hanford Site, and the 100 K and 200 East Areas. Annex A contains information specific to the CSB in the 200 East Area. See Chapters 10 of the facility annexes to the SNF Project Final Safety Analysis Report (FSAR) for specific descriptions of facility sites. This information supports the hazard analysis and accident analyses in Chapters 30 in the facility FSAR Annexes.

12 REQUIREMENTS

The requirements that form the basis for facility siting are found in HNF-SD-SNF-RD-001, Spent Nuclear Fuel Project Standards/Requirements Identification Document. Specific requirements applicable to this chapter include:

- DOE Order 6430 1A, General Design Criteria
- DOE Order 5480 28, Natural Phenomena Hazards Mitigation
- DOE-STD-1022-94, 1994, Natural Phenomena Hazards Site Characterization Criteria
- DOE-STD-1023-95, 1995 Natural Phenomena Hazards Assessment Criteria
- DOE-STD-3014-96, 1996, Accident Analysis for Aircraft Crash into Hazardous Facilities
- NFPA 780 1995, Lightning Protection Systems
In Memorandum EM-36-3 167, *Concurrence with K-Basins Spent Nuclear Fuel Project Policy on Nuclear Safety Requirements* (Grumly 1995), DOE established the requirement for new SNF Project facilities to achieve "nuclear safety equivalency" to comparable U.S. Nuclear Regulatory Commission (NRC)-licensed facilities. The SNF Project identified the NRC requirements that were needed in addition to existing and applicable DOE requirements to establish nuclear safety equivalency. These NRC requirements and the process used to identify them are documented in HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward Additional NRC Requirements*, and in WHC-SD-SNF-DB-010, *Cold Vacuum Drying System Natural Phenomena Hazards*, for CVDF and WHC-SD-SNF-DB-009 *Canister Storage Building Natural Phenomena Hazards*, for CSB.

13 SITE DESCRIPTION

The following sections address the geography, demography, and regional land and water use of the area encompassed by and surrounding the Hanford Site. Much of the information contained in these sections was obtained from DOE/EIS-0245F, *Management of Spent Nuclear Fuel from the K Basins at the Hanford Site Richland Washington*.

13.1 Geography

The Hanford Site is a 560 mi² area located in the state of Washington as shown in Figure 1-1. Principal cities and towns and prominent natural features are shown in Figure 1-1. Use of the Site is institutionally controlled by DOE for national security and health and safety reasons.

The Columbia River enters the Hanford Site at the northwest corner and flows through the northern portion of the Site where it is approximately 1,500 ft from the K Basins and the CVDF (Figure 1-2) before turning southward to form part of the eastern boundary. The CSB, which is in the 200 East Area, is approximately 6.5 mi from the Columbia River on a relatively flat terrace known as the 200 Area Plateau (Figure 1-3). The Yakima River flows from west to east, forming part of the southern boundary of the Site and empties into the Columbia River at the Tri-Cities (Richland, Kennewick, and Pasco) (Figure 1-4). The Hanford Site is bordered on the north by the Saddle Mountains and on the west by the Umtanum and Yakima Ridges and Rattlesnake Mountain (Figure 1-5). Dominant natural features include the Columbia River anticlinal ridges of basalt in and along the Hanford Site boundary, and sand dunes located near the Columbia River. The surrounding basaltic ridges rise to elevations as high as 3,600 ft.

The location of the Hanford Site with respect to local counties and regional highways is shown in Figure 1-4. The Hanford onsite road network consists primarily of rural arterial routes (Figure 1-3). Only 65 of the 288 mi of paved roads are accessible to the public. Most onsite employee travel occurs along Route 4 South and Route 11A, with controlled access at the Yakima and Wye Barricades. An additional access point to the 200 Areas from State Highway 240 (Rattlesnake Barricade) with limited hours of operation is located near the.
southeastern corner of the 200 West Area (see Figure 1-3). Route 11A, the closest to the 100 K Area of the two routes, passes within 4.5 mi of the CVDF and K Basins. Route 4 North, which has less usage, passes within 2.4 mi of the CVDF and K Basins. State Route 240 is the main public route through the Site. State Routes 24 and 243 also traverse the Site. State Route 240 passes through the Hanford Site and within 6 mi of the 100 K Area. Public access through the Hanford Site on Highways 24, 240, and 243 is not strictly controlled by DOE under normal circumstances. Large shipments, in particular components from dismantled nuclear submarines, are offloaded at the Port of Benton dock facilities discussed below. Overland wheeled trailers are then used to transport the shipments to the east side of the 200 East Area. Using this route, the shipments do not pass near the K Basins or CVDF in the 100 K Area or the CSB in the west side of the 200 East Area.

An on-site railroad system is owned by DOE, however, it is not currently in use. This line connects just south of the Yakima River with the Union Pacific line which in turn interchanges with the Washington Central and Burlington Northern–Santa Fe railroads at Kennewick.

There is no barge traffic on the Columbia River upstream (north) of the Port of Benton barge slip which is just north of Richland. The barge slip is located near the upper end of the McNary Dam impoundment and above this location the river is too swift and shallow to allow for safe operation of barges. In addition, there are no lock facilities at Priest Rapids Dam, the next upstream dam and there are no industrial facilities between the Port of Benton barge slip and the dam that would benefit from barge service.

There are two commercial airports within 35 mi of the 100 K Area that serve commercial or military aircraft. These are the Tri-Cities and Richland airports. Besides these, there are a number of small commercial and privately owned airports exclusively serving general aviation aircraft within a 24 mi radius of the 100 K Area and 200 East Area. They are discussed in detail in Section 1.6.1.

Traffic on the Columbia River in the airspace over the Hanford Site and on access routes into the areas used by U.S. Ecology Incorporated and Energy Northwest (formerly the Washington Public Power System) are not subject to strict controls under normal circumstances (see Figure 1-3). Under emergency plan conditions, DOE will control all road and railroad access to the Hanford Site and may close these Site routes to normal traffic. Notification of offsite agencies in the event of an emergency is provided so that they may take measures within their jurisdictions to ensure public protection (see Chapter 15). The U.S. Coast Guard and the Federal Aviation Administration in cooperation with DOE and other agencies, would assess the nature of the emergency and restrict river and airspace traffic if such action were deemed prudent for public safety.

**1311 Hanford Site Vegetation** The most broadly distributed vegetation on the Hanford Site is sagebrush, wheatgrass, blue bunch wheatgrass, cheatgrass, and other shrub plant species common to central Washington. The flora is prone to destruction by fire and can burn rapidly. The most recent severe grass fire at the Hanford Site occurred in August 1984 as a result of a range fire that started northeast of Sunnyside, Washington (see Figure 1-1). Extremely dry
vegetation, inaccessible mountain terrain, and winds up to 40 m/h limited attempts to extinguish
the fire. The fire front entered the Hanford Site adjacent to Rattlesnake Mountain, crossed
Highway 240, and traveled northeast along Army Loop Road south of the 200 East Area
(Figure 1-3) The fire was not controlled until it reached the Columbia River, by which time it
had burned approximately 312 mi² of range land. Hanford Site facilities near the fire were
protected by fire breaks cut by bulldozers and blades, and by fire-fighting tanker trucks. Vehicle
access is available for the areas around all major Hanford Site facilities. For this fire, air drops of
fire retardant material by U.S. Forest Service planes from the Wenatchee National Forest were
used to help protect major facilities. Losses related to this severe grass fire were limited to
burned railroad ties, wooden power poles, and several trailers. Several fire-fighting vehicles also
were damaged.

To protect the SNF Project facilities, a clear space is provided on all sides of the facility.
Unpaved areas are overlaid with a crushed rock surface. The potential impact of range fires on
SNF Project operation is addressed in the facility-specific hazard analyses.

13.1.2 Hanford Site Facilities In 1943 the U.S. Army Corps of Engineers selected the
Hanford Site for construction of nuclear reactors and chemical processing facilities in support of
the war effort. The current mission is environmental management of radioactive and hazardous
wastes, restoration of Site land, and conversion of useable facilities for other uses. DOE nuclear
facilities currently occupy approximately 6% of the total available Hanford Site land area. Site
operating areas are identified by area numbers and are briefly described below. Figure 1-3 shows
these numbered areas.

Six 100 Areas border directly on the Columbia River in the northernmost portion of the
Site. These areas were the locations for nine graphite-moderated plutonium production reactors.
Eight of these reactors, all of which started operation before 1960, were shut down in the
early 1970s. The ninth reactor, N Reactor, became operational in 1963 and continued operating
until 1986. The K Basins and the CVDF are located in the 100 K Area. The 100 Areas occupy
about 4 mi².

The 200 East and 200 West Areas are located near the center of the Hanford Site on a
relatively flat terrace known as the 200 Area Plateau. In the past, the 200 East and 200 West
Areas have received waste from the 100, 300, and 400 Areas in addition to the waste produced by
200 Area separation processes. The CSB is located in the 200 East Area. The 200 Areas occupy
approximately 8 mi².

The DOE has awarded a contract to a private company to design facilities on the east side
of the 200 East Area for the vitrification of the Hanford Site tank farms waste. At the conclusion
of the design phase, DOE will decide whether to proceed into a construction and operation phase.
These facilities will be located near the 241-AP Tank Farm. The effect of these and other new
facilities on CSB operations will be evaluated in scheduled updates of the SNF Project FSAR as
planning, construction, and operation proceed.
The 300 Area, located just north of Richland, is the site of various nuclear research and development facilities. This area covers 0.6 mi².

The 400 Area is the site of the Fast Flux Test Facility, a liquid-metal-cooled fast reactor used for testing breeder reactor fuels, materials, and components. The Fast Flux Test Facility is currently defueled and in standby mode, awaiting decision by the DOE to restart for future missions or proceed with preparations for eventual decontamination and decommissioning activities. The 400 Area covers 0.25 mi².

The 600 Area includes all of the Hanford Site not occupied by the 100, 200, 300, or 400 Areas. Land uses within the 600 Area include the following (Figure 1-3):

- An area measuring 120 mi² known as the Fitzner-Eberhardt Arid Lands Ecology Reserve, is located along the southwest boundary of the Hanford Site (Figure 1-5). This area is a limited-access area and has been set aside for ecological studies. The DOE may in the future enter into an agreement with the U.S. Fish and Wildlife Service to manage the area.

- An area measuring 15 mi², located between the 200 West and 200 East Areas, is leased by Washington State. A part of this area is managed by U.S. Ecology, Incorporated for the disposal of solid, low-level radioactive waste from commercial— not Hanford Site — sources.

- An area measuring 16 mi² is used by Energy Northwest for commercial nuclear power plants. This area is located approximately 10.5 mi southeast of the 200 East Area and approximately 3 mi west of the Columbia River. Three commercial nuclear power plants were originally planned for this site. Washington Nuclear Plant (WNP)-2, a boiling-water reactor, is currently in operation at this location. Construction has been terminated at the other two reactors, WNP-1 and WNP-4. Both were to be pressurized water reactors. Also located on the Energy Northwest leased property are the Plant Engineering Center and Plant Support Facility and the Bonneville Power Administration's H.J. Ashe substation.

- An area measuring 1 mi² has been transferred to Washington State as a potential site for the disposal of nonradioactive hazardous wastes. See Figure 1-3.

- About 50 mi² of land north of the Columbia River, the Saddle Mountains National Wildlife Refuge, is managed under a revocable use permit by the U.S. Fish and Wildlife Service. The U.S. Fish and Wildlife Service also manages seven islands in the Hanford Reach section of the Columbia River as part of the McNary National Wildlife Refuge.

- Another 87 mi² of land north of the Columbia River, the Wahluke Wildlife Recreation Area, is under a revocable use permit managed by the Washington State Department of Wildlife for recreational game management. Public outdoor recreational use is permitted in this area.
An area of less than 1 m² is occupied by the 616 Nonradioactive Dangerous Waste Disposal Facility. It is located approximately 200 ft north of Route 3 between the 200 East and 200 West areas. In the past, the facility received, stored, and shipped nonradioactive waste designated as extremely hazardous and dangerous that was generated at all DOE, Richland Operations Office facilities.

Support facilities for the controlled access areas and an electrical transmission substation at Midway near the northwest corner of the Hanford Site are maintained by the Bonneville Power Administration. The Hanford Site electrical distribution system (i.e., transmission lines and substations) are discussed in Section 2.8.

1313 Boundaries for Evaluation of Accident and Effluent Release Limits Activities in the 100 K Area and 200 East Area are within the DOE-controlled zone. A security fence limits general access by land. The fence stops at the high-water mark of the Columbia River. Access from the river is controlled by "No Trespassing" signs. The Hanford Patrol is responsible for enforcing the restriction on access to the area, including access from the river. No special security clearance other than a Site badge is required to pass through the gates that control access to the 100 K Area and 200 East Area. However, the DOE has the authority to determine all activities in this area including exclusion or removal of personnel and property. There are no permanent residences in this area. Specific boundaries used in the accident analyses are described in the facility FSAR Annexes.

1314 Regional Land and Water Use This section characterizes regional land and water use both inside and outside the Hanford Site boundaries based on information contained in DOE/EIS-0113, Final Environmental Impact Statement Disposal of Hanford Defense High-Level Transuranic and Tank Wastes Hanford Site, Richland Washington, PNL-6415, Hanford Site National Environmental Policy Act (NEPA) Characterization, and DOE/EIS-0245F.

13141 Land Use Within the Hanford Site Boundaries The majority of the land within the Hanford Site boundary is a limited-access area under control of the DOE for use in environmental restoration and remediation efforts. A number of areas at the Hanford Site are managed under a multipurpose concept and serve as buffer zones around areas of nuclear activity. These multipurpose areas are shown in Figure 1-3.

DOE is in the process of preparing a Hanford Site land use plan to comply with state and county land use planning laws and regulations. This plan establishes a blueprint for use of Hanford Site lands consistent with the Site mission of environmental cleanup and radioactive waste management. The strategy is documented in the revised DOE/EIS-0222D Draft Hanford Remedial Action Environmental Impact Statement and Comprehensive Land Use Plan (DOE 1998) currently being reviewed by DOE.

13142 Agricultural Land Use Outside Hanford Site Boundaries Land use in the six-county region surrounding the Hanford Site (i.e., Franklin, Walla Walla, Benton, Yakima, Adams, and Grant Counties [Figure 1-4]) is predominantly agricultural. Over 75% of the land...
area in this region is in agriculture-related use compared with less than 40% statewide. The percentage of agricultural land that is irrigated in this region varies from a high of 73% in Grant County to a low of 33% in Yakima County. However, areas of Yakima County near the Hanford Site have intensive irrigation.

Regional land used for agricultural purposes lies primarily north and east of the Columbia River and south of the Yakima River. The U.S. Bureau of Reclamation's Columbia Basin Irrigation Project is northeast of Othello. Part of the farmland northeast of Priest Rapids Dam is irrigated by the Columbia Basin Project; the rest is used for dryland farming. The area from Othello south to the Columbia River is irrigated farmland. The principal agricultural crops associated with the irrigated and dryland farming include hay, wheat, potatoes, corn, apples, soft fruit, and vegetables. Land in the area just outside the Hanford Site to the south and west is available for farming but is used mainly for grazing cattle and sheep. The agricultural land along the Yakima River is used for fruit orchards, grapes, hops, alfalfa, and annual row crops. Higher land away from the Yakima Valley floor is used principally for livestock grazing, dairies, and fruit orchards; with some grain crops. South and west of Kennewick, the land is used principally for dryland wheat farming, with some orchards.

13143 Nonagricultural Land Use Outside Hanford Site Boundaries The main industries in the Tri-Cities are, or are related to, agriculture and energy production. The DOE's nuclear and nonnuclear industrial activities within a 5-mile radius of the SNF Project facilities are described in Sections 17 of the facility FSAR Annexes. All industrial activities on Hanford Site leased land must be compatible with DOE activities and must be approved by the DOE. These land uses are discussed in Section 1312. The DOE retains the right to remove from lease any land not currently used or not under development by the state. The Big Bend Alberta Company owns mineral rights to several parcels of land in the Arid Lands Ecology Reserve (Figure 1-3) and holds the right to perform exploratory drilling.

The U.S. Army's Yakima Training Center covers 326,000 acres just northwest of the western Hanford Site boundary (Figure 1-4). The Center is used for firing all types of ordnance both in a direct mode and by indirect artillery and mortars. Weapons to 8-in are fired. Firing occurs frequently. Live ordnance used at the Center includes bombs up to 500 lb delivered by high-performance aircraft, helicopter weapons, which include automatic weapons and 2.75-in folding fin rockets and anti-aircraft missiles. The majority of the ordnance impacts a 20,000-acre area that is located in the central portion of the Center. All activities are confined to the geographical limits of the Center and/or its restricted air space unless special arrangements are made with affected agencies. Mechanized units (i.e., tanks and armored personnel carriers) from Fort Lewis and Reserve components conduct extensive maneuvers on all accessible areas of the Yakima Training Center and use specially designed ranges to practice firing their weapons. Training activity is greatest from March to November. War games sometimes involve troop and equipment deployment at the Richland airport and along Highway 243 west of the Vernita Bridge. Helicopters may fly near the Hanford Site, or military vehicles may travel over Highway 240.

13144 Water Use The major water uses, in decreasing order, are agricultural, industrial, and municipal. CVDF and CSB operation will involve no water withdrawals from or
direct discharges to the Columbia River Water removed from the multi-canister overpacks at the CVDF will be returned to K West Basin See WHC-SD-WM-SAR-062 K Basins Safety Analysis Report, for a description of the K Basins interface with the Columbia River

132 Demography

1321 Offsite Population The 1990 U.S. Bureau of the Census population distribution statistics for cities within a 50-mi radius of the Hanford Meteorological Station (HMS) are shown in Figure 1-6.

Population estimates for 1993 by the Forecasting Division of the Office of Financial Management of the State of Washington place the totals for Benton, Franklin, and Grant counties at 122,800, 41,100, and 60,300, respectively. The 1993 estimates for the Tri-Cities populations are Richland 34,080, Kennewick, 45,220, and Pasco 21,370. The estimated populations of Benton City, Prosser, and West Richland totaled 10,900 in 1993 (PNL-9823).

Major metropolitan areas within the broad vicinity of the SNF Project facilities include Spokane, Washington, about 120 mi to the northeast, Seattle, Washington, about 130 mi to the northwest and Portland, Oregon, about 150 mi to the southwest. Two other areas of significant population density include Moses Lake, Washington, about 30 mi north of the 100 K Area and the Yakima Valley, in Washington, extending from Yakima, about 45 mi west of the 100 K Area, to the Tri-Cities, in Washington, about 35 mi southeast of the plant.

The population distribution in the area surrounding the Hanford Site is not uniform. Most of the adjacent area to the east, north, and west is farm or range land with scattered farming communities. Table 1-1 shows the actual 1990 residential population and the projected residential population within a 50-mi radius of the HMS for decennial census years through 2040 (PNL-7803).

<table>
<thead>
<tr>
<th>Year</th>
<th>Decade</th>
<th>Average annual rate of growth during decade (%)</th>
<th>Population</th>
</tr>
</thead>
<tbody>
<tr>
<td>1990</td>
<td>--</td>
<td>--</td>
<td>375,860</td>
</tr>
<tr>
<td>2000</td>
<td>1990 to 2000</td>
<td>0.633</td>
<td>400,346</td>
</tr>
<tr>
<td>2010</td>
<td>2000 to 2010</td>
<td>0.413</td>
<td>417,200</td>
</tr>
<tr>
<td>2020</td>
<td>2010 to 2020</td>
<td>0.351</td>
<td>432,062</td>
</tr>
<tr>
<td>2030</td>
<td>2020 to 2030</td>
<td>0.157</td>
<td>438,909</td>
</tr>
<tr>
<td>2040</td>
<td>2030 to 2040</td>
<td>0.068</td>
<td>441,911</td>
</tr>
</tbody>
</table>
The Tri-Cities are located to the south-southeast of the 100 K Area and comprise the major population center of the area. The three cities (Kennewick, Pasco, and Richland) are estimated to have a combined population of approximately 104,000 based on 1994 estimates. The estimated unincorporated population of Benton County is 33,000 and of Franklin County is 18,000 (DOE/EIS-0245F). More recent data on the Tri-Cities population are not available although Hanford Site and total nonfarm employment are on a downward trend. The projected increase in the total population of Franklin County from 1995 to 2005 is from 41,336 to 48,213 (8.03%) and for Benton County from 121,328 to 136,892 (6.32%) (DOE 1995). The nearest residence to the 200 East Area is approximately 11 mi east across the Columbia River. The Richland city limits are approximately 16 mi to the southeast.

Other population centers of note within a 50-mi radius include the cities of Othello 22 mi east-northeast Mattawa 17 mi west-northwest and Yakima 43 mi west. The Yakima River Valley, stretching in an arc from the city of Yakima to the Tri-Cities, is a relatively densely populated agricultural area with a number of small towns.

There are currently no schools, hospitals, nursing homes, or penal institutions within 12 mi of the SNF Project facilities. The closest schools are located at Mattawa 17 mi west-northwest. These Mattawa schools, and their 1995 enrollments are:

- Mattawa Elementary School 482 students
- Morris Scott Middle School 301 students
- Wahluke High School 232 students

Also located in Mattawa is a medical clinic. There are currently no hospitals, nursing homes, or penal institutions in Energy Northwest's 10-mi emergency planning zone. The closest school to the CSB site is the Edwin Markham Elementary School with a 1995-1996 enrollment of 283 students. Evacuation of the Edwin Markham School and of the nearby Country Haven and Country Christen School is provided for in the Benton and Franklin Counties Fixed Nuclear Facility Emergency Response Plan.

1322 Onsite Population In 1996 approximately 11,000 persons were employed on the Hanford Site. Estimated employee populations located near the SNF Project facilities are:

- 305 at the 100 K Area
- 10 at the 100 D Area
- 379 at the 100 N Area
- 2,750 at the 200 East Area

Some Hanford Site work assignments include shift and weekend coverage. The total number of persons on the Site at any one time varies with the time of day, staffing requirements for current projects, and daily fluctuations in employee work attendance.

1323 Transient Population Figure 1-7 shows the estimated populations of migrant agricultural workers and recreationists. Both are defined as transient. These estimates were
developed by the then Washington Public Power Supply System in 1988 and are documented in *WNP-I, 2 Ten Mile EPZ Evacuation Time Assessment Study* (Mogle 1987). The center of the 10-m emergency planning zone, which is located midway between WNP-1, WNP-2, and WNP-4 is the geographic point from which estimates are made. The CSB is located approximately 12 m west-northwest from the center point. 100 K Area is located approximately 20 m northwest of the centerpoint (see Figure 1-7). This distance from the center point is not significant to the accident analyses in the facility FSAR Annexes as the location of the offsite receptor is determined by consideration of the Site boundary and the ability to control activities within the boundary as discussed in Section 13.1.3.

### 13.2.4 Emergency Response Considerations

Figure 1-3 shows the major roads that would be used for onsite and offsite evacuation routes. The roads shown will provide adequate evacuation of collocated workers and the general public, assuming worst-case evacuation conditions and maximum evacuation rates. Access to designated shelter or shielding areas is provided by the road system for worst-case evacuation rates considering anticipated delays. Sheltering and shielding parameters are included in the facility building emergency plans described in Section 15.4. Facility-specific onsite and offsite emergency response plans will address specific actions (i.e., notifications, take-cover, facility shutdown, evacuation) that are dependent upon the emergency condition. See Section 15.4 for a discussion of how the emergency response plans are developed.

### 14 Environmental Description

The following sections address the meteorology, hydrology, and geology of the Hanford Site.

#### 14.1 Meteorology

This section describes the regional climatology and the meteorological conditions of the Hanford Site. Climatological data are available for the HMS, which is located between the 200 East and 200 West Areas, approximately 6 mi south of the 100 K Area (Figure 1-8). Data have been collected at this location since 1945. Data from the HMS are representative of the general climatic conditions for the region and describe the specific climate of the 200 Area Plateau. Information contained in this section is taken primarily from PNL-11794 *Climatological Data Summary 1997 with Historical Data*, and PNL-4622 *Climatological Summary for the Hanford Area*. These data are used in the preparation of safety documentation and in the development of emergency response planning.

#### 14.1.1 Regional Meteorology

A wide range of meteorological variables is measured at the HMS and at the 125-m tower, which is located 1,614 ft east of the HMS. Temperature, relative humidity, precipitation, atmospheric pressure, solar radiation, cloud cover, visibility, and subsurface temperature are parameters measured or observed at the HMS. Wind data are measured at various levels on the 125-m tower. Three 200-ft towers with wind and temperature
measuring instruments at various levels, are located at the 300, 400, and 100 N Areas. Wind and temperature measurements also are taken on 23 30-ft towers distributed on and around the Hanford Site (see Figure 1-8). Data from all towers are telemetered to the HMS. The Hanford Meteorological Monitoring Network is described in detail in PNL-6684, *The Data Collection Component of the Hanford Meteorological Monitoring Program*.

Meteorological parameters have been measured in the vicinity of the Hanford Site since 1912. The HMS became operational in December of 1944. Since 1944, additional large-scale meteorological monitoring towers have also been constructed, including one in the 100 N Area.

**1412 Regional Climatology** The climate of the Pasco Basin, in which the 100 N Area is located, can be classified as midlatitude semiarid or midlatitude desert, depending on the climatological classification scheme used. Summers are warm and dry with abundant sunshine. Large diurnal temperature variation results from intense solar heating during the day and radiational cooling at night. Daytime high temperatures in June, July, and August periodically exceed 100 °F. Winters are cool with occasional precipitation. Outbreaks of cold air associated with modified arctic air masses can reach the area and cause temperatures to drop below 0 °F. Overcast skies and fog occur periodically during the winter season (PNL-4622).

Topographic features have a significant impact on the climate of the Hanford Site. All air masses that reach the Pasco Basin undergo some modification as a result of their passage over the complex terrain of the Pacific Northwest. The climate of the Pasco Basin is strongly influenced by the Pacific Ocean and the Cascade Range to the west. The relatively low annual average rainfall of 6.8 in. at the HMS is caused largely by the rain shadow created by the Cascade Range. This mountain range limits much of the maritime influence of the Pacific Ocean, resulting in a more continental-type climate than would exist if the range were not present. Maritime influences are experienced in the region during the passage of frontal systems and as a result of movement through the gaps in the Cascade Range (e.g., Columbia River Gorge).

The Rocky Mountains to the east and north of the Pasco Basin also influence the climate of the region. These mountains play a key role in protecting the region from the more severe winter storms and the extremely low temperatures associated with modified arctic air masses that move southward through Canada. The Yakima Ridge, Rattlesnake Hills, and Horse Heaven Hills to the west and south of the Hanford Site, the Saddle Mountains to the north, and the Columbia River also influence the local climate (see Figure 1-9).

The position of the jet stream directs storm systems into Washington State during the cooler months. These systems cause the majority of the precipitation at the Hanford Site. The cold, or occluded, fronts associated with these storm systems originate in maritime polar, continental polar, or arctic air masses. Warm fronts occur as warmer maritime air flows over colder continental air. An average of 10 identifiable warm fronts and 52 cold fronts pass through the Pasco Basin each year (PNL-4622). A majority of these passages occur during the colder months leading to increased frequency and intensity of precipitation. Periodic persistent high-pressure ridges in the upper troposphere in the cool season, which can trap cold air near the surface, lead to extended periods of wintertime stagnation.
The jet stream directs most storms north of the state of Washington during the warmer months, therefore frontal passages are fewer and weaker than in the cooler months. High pressure with stable, subsiding air is the dominant meteorological condition during this period. Warm weather precipitation tends to be associated with convective activity and the advection of moist, maritime air into the region (PNL-4622)

14121 Temperature Monthly and annual daily average temperatures for the Hanford Site from 1961 through 1990 and extreme temperatures from 1945 through 1997 are given in Table 1-2. At the Hanford Site, the annual average temperature is 53 °F (PNL-11794). July is typically the warmest month with an average maximum temperature of 91 °F, an average minimum temperature of 61 °F, and an average temperature of 76 °F. January tends to be the coolest month with an average maximum temperature of 38 °F, an average minimum temperature of 24 °F, and an average temperature of 32 °F. Observed temperature extremes for the Hanford Site range from 113 °F to -23 °F. The highest temperature ever recorded on the Hanford Site was 115 °F on July 27, 1939. The lowest temperature ever recorded on the Hanford Site was -27 °F on December 12, 1919. These temperatures were recorded before the HMS was operational and are not included in Table 1-2.

<table>
<thead>
<tr>
<th>Month</th>
<th>Average daily temperature, 1961-1990 (°F)</th>
<th>Extreme temperature, 1945-1997 (°F)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Maximum</td>
<td>Minimum</td>
</tr>
<tr>
<td>January</td>
<td>38</td>
<td>24</td>
</tr>
<tr>
<td>February</td>
<td>47</td>
<td>29</td>
</tr>
<tr>
<td>March</td>
<td>57</td>
<td>34</td>
</tr>
<tr>
<td>April</td>
<td>66</td>
<td>40</td>
</tr>
<tr>
<td>May</td>
<td>75</td>
<td>47</td>
</tr>
<tr>
<td>June</td>
<td>84</td>
<td>55</td>
</tr>
<tr>
<td>July</td>
<td>91</td>
<td>61</td>
</tr>
<tr>
<td>August</td>
<td>90</td>
<td>60</td>
</tr>
<tr>
<td>September</td>
<td>80</td>
<td>51</td>
</tr>
<tr>
<td>October</td>
<td>66</td>
<td>40</td>
</tr>
<tr>
<td>November</td>
<td>49</td>
<td>32</td>
</tr>
<tr>
<td>December</td>
<td>38</td>
<td>25</td>
</tr>
<tr>
<td>Year</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Note: Information in this table is from PNL 11794 1998 Climatological Data Summary 1997 with Historical Data Pacific Northwest National Laboratory Richland Washington
Table 1-3 shows the number of days per month averaged over the period 1945 through 1997, that the Hanford Site maximum and minimum temperatures were above or below specific values (PNL-11794) Maximum temperatures greater than or equal to 100 °F occur an average of 11 days a year and vary from 1 to 28 days Maximum temperatures greater than or equal to 90 °F occur an average of 51 days per year and vary from a low of 29 to a high of 79 days Maximum temperatures less than or equal to 32 °F occur an average of 25 days a year and vary from 2 to 58 days Minimum temperatures less than or equal to 32 °F occur an average of 107 days per year and vary from a low of 70 to a high of 143 days Minimum temperatures less than or equal to 0 °F occur an average of 3 days per year and vary from 0 to 18 days An average of 183 days a year are free of freezing temperatures, with the recorded range being 142 to 216 days

Table 1-3  Average Number of Days at the Hanford Meteorological Station that Maximum and Minimum Temperatures were above or below Specific Limits for the Period 1945 through 1997

<table>
<thead>
<tr>
<th>Month</th>
<th>Maximum temperature</th>
<th>Minimum temperature</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>≥100 °F</td>
<td>≥90 °F</td>
</tr>
<tr>
<td>January</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>February</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>March</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>April</td>
<td>0</td>
<td>*</td>
</tr>
<tr>
<td>May</td>
<td>*</td>
<td>3</td>
</tr>
<tr>
<td>June</td>
<td>1</td>
<td>8</td>
</tr>
<tr>
<td>July</td>
<td>6</td>
<td>19</td>
</tr>
<tr>
<td>August</td>
<td>4</td>
<td>16</td>
</tr>
<tr>
<td>September</td>
<td>*</td>
<td>6</td>
</tr>
<tr>
<td>October</td>
<td>0</td>
<td>*</td>
</tr>
<tr>
<td>November</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>December</td>
<td>0</td>
<td>0</td>
</tr>
<tr>
<td>Year</td>
<td>12</td>
<td>52</td>
</tr>
</tbody>
</table>

Note  Information in this table is from PNL 11794  1998 Climatological Data Summary 1997 with Historical Data  Pacific Northwest National Laboratory Richland Washington

*Denotes less than half a day
The Hanford Site and vicinity are known for severe and abrupt temperature changes. During winter the Site frequently experiences rapid rises in temperature accompanied by moderate west winds. This phenomenon, known as a chinook wind, has produced temperature variations of up to 22 °F in 0.5 hours and is responsible for rapid melting of snow (DOE/EIS-0113).

**14122 Precipitation** The annual average precipitation value at the HMS is 6.8 in with the annual precipitation value for the wettest documented year (1995) being 12.3 in and for the driest year (1976) being 2.99 in (PNL-11794). Monthly average and extreme precipitation amounts for the Hanford Site from 1945 through 1997 are given in Table 1-4. On average, 54% of normal annual precipitation falls during November through February. December is the wettest month receiving 1.03 in, and July is the driest month receiving only 0.18 in. The wettest month on record is December 1996 with 3.7 in, September 1991, August 1988, and August 1955 recorded no precipitation. An average of 125 days per year have a trace (less than 0.005 in) or more of precipitation. The average number of days per month with a trace or more ranges from 16 days in January to 5 days in July. Only 23 days a year receive totals of 0.1 in or more. During the 51-year period of record (1945 through 1997) only 4 days have had 1 in or more of precipitation.

<table>
<thead>
<tr>
<th>Month</th>
<th>Average</th>
<th>Maximum</th>
<th>Year</th>
<th>Minimum</th>
<th>Year</th>
</tr>
</thead>
<tbody>
<tr>
<td>January</td>
<td>0.92</td>
<td>2.47</td>
<td>1970</td>
<td>0.08</td>
<td>1977</td>
</tr>
<tr>
<td>February</td>
<td>0.62</td>
<td>2.10</td>
<td>1961</td>
<td>T</td>
<td>1988</td>
</tr>
<tr>
<td>March</td>
<td>0.51</td>
<td>1.86</td>
<td>1957</td>
<td>0.02</td>
<td>1968</td>
</tr>
<tr>
<td>April</td>
<td>0.45</td>
<td>1.54</td>
<td>1995</td>
<td>T</td>
<td>1986</td>
</tr>
<tr>
<td>May</td>
<td>0.53</td>
<td>2.03</td>
<td>1972</td>
<td>T</td>
<td>1992</td>
</tr>
<tr>
<td>June</td>
<td>0.53</td>
<td>2.92</td>
<td>1950</td>
<td>T*</td>
<td>1986</td>
</tr>
<tr>
<td>July</td>
<td>0.22</td>
<td>1.76</td>
<td>1993</td>
<td>T</td>
<td>1980</td>
</tr>
<tr>
<td>August</td>
<td>0.24</td>
<td>1.36</td>
<td>1977</td>
<td>0*</td>
<td>1988</td>
</tr>
<tr>
<td>September</td>
<td>0.32</td>
<td>1.34</td>
<td>1947</td>
<td>0</td>
<td>1991</td>
</tr>
<tr>
<td>October</td>
<td>0.55</td>
<td>2.72</td>
<td>1957</td>
<td>T</td>
<td>1987</td>
</tr>
<tr>
<td>November</td>
<td>0.90</td>
<td>2.67</td>
<td>1996</td>
<td>T</td>
<td>1976</td>
</tr>
<tr>
<td>December</td>
<td>1.02</td>
<td>3.69</td>
<td>1996</td>
<td>0.11</td>
<td>1976</td>
</tr>
</tbody>
</table>

Notes: Information in this table is from PNL 11794 1998 Climatological Data Summary 1997 with Historical Data Pacific Northwest National Laboratory Richland Washington.

*Most recent of multiple occurrences.

T = trace (0.005 in or less)
Hydrometeorological Report No 57, Probable Maximum Precipitation — Pacific Northwest States (Hansen et al 1994), a recent cooperative study by the National Oceanic and Atmospheric Administration, the Bureau of Reclamation and the U.S. Army Corps of Engineers, has updated the probable maximum precipitation (PMP) estimates for the Pacific Northwest. This document supersedes earlier work done by these organizations and is the source used for the PMP shown in Table 1-5. The PMP values are estimates of the maximum precipitation physically possible for both general storms (large air mass interactions) and local storms (unstable air thunderstorms). At the Hanford Site, the 6-hour local storm produces more precipitation than the 24-hour general storm. The 6-hour local storm PMP is related to the area of the storm, the smaller area yielding the most intense storm and highest precipitation. Data are presented for the 1-m$^2$ and the 10-m$^2$ storm. The SNF Project facilities are designed to accommodate the full range of PMP identified in Table 1-5. The effects of local PMP are determined according to ANSI/ANS-2 8-1992. Local protection for run-off is provided by earthwork and grading design and by the storm drain system consisting of culverts, ditches, channels, and catch basins as required.

No annual probability of exceedance is given in Hydrometeorological Report No 57 (Hansen et al 1994) for the PMP for either general or local storms. The PMP is conservatively assumed to have an annual probability of exceedance of less than $1 \times 10^{-6}$ (ASCE 1988).

The 6-hour PMP for more frequent storms shown in Table 1-5 is from PNL-4622 and is based on the analysis of extreme values from 22 years of meteorological data from the Hanford Site. The precipitation estimates for the 100- and 1,000-year return periods are based on data from the HMS. Although these values cannot be compared directly with either the 1-m$^2$ storm or the 10-m$^2$ storm, they provide a data-based estimate for extreme precipitation on the 200 Area Plateau. A 6-hour precipitation hazard curve is estimated using the 100-year and 1,000-year average return period values ($10^2$ and $10^3$ annual probability of exceedance respectively) of PNL-4622 and the 6-hour PMP at an assumed frequency of $10^{-6}$ (Figure 1-10).

Total annual snowfall, which includes all frozen precipitation, varied for the period 1945 through 1997 from a low of 0.3 in to a high of 56.1 in (PNL-11794). The average snowfall is 15.3 in a year. Table 1-6 presents the monthly and annual average and extreme snowfalls. The record snow depth at the HMS is 15.6 in in December 1995, but the record snow depth on the Hanford Site is 24 in in February 1916. The record number of days with snow depth greater than or equal to 6 in is 43 days in the winter of 1992-1993. The probability the depth of snow on the ground will exceed 6 in is 30% during the winter season (Figure 1-11).

The design ground snow load for the CVDF and CSB is 20 lb/ft$^2$ which is the ground snow load for the NRC-licensed WNP-2. This value is 5 lb/ft$^2$ greater than the ASCE requirement (ASCE 7-95).
Table 1-5  Extreme Precipitation Estimates for the Hanford Site

<table>
<thead>
<tr>
<th>Time</th>
<th>PMP 24-hour general storm (10 m²)(^a)</th>
<th>PMP local storm (1 m²)(^a)</th>
<th>PMP local storm (10 m²)(^a)</th>
<th>25-year average return period(^b)</th>
<th>100-year average return period(^b)</th>
<th>1 000-year average return period(^b)</th>
</tr>
</thead>
<tbody>
<tr>
<td>15 minutes</td>
<td>--</td>
<td>4 0</td>
<td>3 2</td>
<td>--</td>
<td>--</td>
<td>--</td>
</tr>
<tr>
<td>20 minutes</td>
<td>--</td>
<td>--</td>
<td>--</td>
<td>0 47</td>
<td>0 60</td>
<td>0 80</td>
</tr>
<tr>
<td>30 minutes</td>
<td>--</td>
<td>6 0</td>
<td>4 8</td>
<td>--</td>
<td>--</td>
<td>--</td>
</tr>
<tr>
<td>45 minutes</td>
<td>--</td>
<td>7 2</td>
<td>5 8</td>
<td>--</td>
<td>--</td>
<td>--</td>
</tr>
<tr>
<td>1 hour</td>
<td>1 6</td>
<td>8 0</td>
<td>6 4</td>
<td>0 62</td>
<td>0 81</td>
<td>1 11</td>
</tr>
<tr>
<td>6 hours</td>
<td>4 7</td>
<td>9 2</td>
<td>7 4</td>
<td>1 21</td>
<td>1 59</td>
<td>2 20</td>
</tr>
<tr>
<td>24 hours</td>
<td>8 0</td>
<td>--</td>
<td>--</td>
<td>1 56</td>
<td>1 99</td>
<td>2 68</td>
</tr>
<tr>
<td>48 hours</td>
<td>9 6</td>
<td>--</td>
<td>--</td>
<td>--</td>
<td>--</td>
<td>--</td>
</tr>
<tr>
<td>72 hours</td>
<td>10 4</td>
<td>--</td>
<td>--</td>
<td>--</td>
<td>--</td>
<td>--</td>
</tr>
</tbody>
</table>

Note: Precipitation depths are in inches. To convert to centimeters multiply by 2.54


PMP = probable maximum precipitation.
**Table 1-6** Monthly and Seasonal Snowfall at the Hanford Meteorological Station in Inches, Average and Maximum for the Period 1945 through 1997

<table>
<thead>
<tr>
<th>Month</th>
<th>Average</th>
<th>Maximum</th>
<th>Year</th>
<th>Maximum 24-hour</th>
<th>Year</th>
</tr>
</thead>
<tbody>
<tr>
<td>January</td>
<td>50</td>
<td>23.4</td>
<td>1950</td>
<td>7.1</td>
<td>1954</td>
</tr>
<tr>
<td>February</td>
<td>25</td>
<td>17.0</td>
<td>1989</td>
<td>10.2</td>
<td>1993</td>
</tr>
<tr>
<td>March</td>
<td>0.5</td>
<td>4.2</td>
<td>1951</td>
<td>2.7</td>
<td>1989</td>
</tr>
<tr>
<td>April</td>
<td>1.0</td>
<td></td>
<td>1982</td>
<td></td>
<td></td>
</tr>
<tr>
<td>May</td>
<td>0</td>
<td>0</td>
<td>--</td>
<td>0</td>
<td>--</td>
</tr>
<tr>
<td>June</td>
<td>0</td>
<td>0</td>
<td>--</td>
<td>0</td>
<td>--</td>
</tr>
<tr>
<td>July</td>
<td>0</td>
<td>0</td>
<td>--</td>
<td>0</td>
<td>--</td>
</tr>
<tr>
<td>August</td>
<td>0</td>
<td>0</td>
<td>--</td>
<td>0</td>
<td>--</td>
</tr>
<tr>
<td>September</td>
<td>0</td>
<td>0</td>
<td>--</td>
<td>0</td>
<td>--</td>
</tr>
<tr>
<td>October</td>
<td>0.1</td>
<td>1.5</td>
<td>1973</td>
<td>1.5</td>
<td>1973</td>
</tr>
<tr>
<td>November</td>
<td>1.8</td>
<td>18.3</td>
<td>1985</td>
<td>8.8</td>
<td>1985</td>
</tr>
<tr>
<td>December</td>
<td>5.3</td>
<td>22.6</td>
<td>1996</td>
<td>6.6</td>
<td>1985</td>
</tr>
<tr>
<td>Year</td>
<td>15.3</td>
<td>56.1</td>
<td>1992-93</td>
<td>10.2</td>
<td>February 1993</td>
</tr>
</tbody>
</table>

Notes: Information in this table is from PNL 11794 1998 Climatological Data Summary 1997 with Historical Data Pacific Northwest National Laboratory Richland Washington

*No value given for maximum 24 hour snowfall

Glaze ice is a coating of ice formed when rain or drizzle freezes on contact with any surface having a temperature that is below freezing. The average number of days with freezing rain or freezing drizzle is six and they generally occur between November and March. The highest number of days with glaze in any winter season was 18. The least number of days with glaze in any winter season was one. The greatest number of days with glaze in any given month was nine in January 1970. Rime ice (i.e., supercooled droplets that freeze on contact with solid objects) is generally associated with supercooled fog in the nearby hills or along the banks of the Columbia River.

14123 Thunderstorms A thunderstorm day is one in which thunder is heard at the observing station one or more times during a calendar day. Table 1-7 shows the number of thunderstorm days per month and per year for the period 1945 through 1997. The average number of thunderstorm days per year is 10. The total varies from a low of 3 to a high of 23 days (PNL-11794). The largest number of thunderstorm days in a single month was eight.
Table 1-7  Average Number of Thunderstorm Days at the Hanford Meteorological Station for the Period 1945 through 1997

<table>
<thead>
<tr>
<th>Month</th>
<th>Thunderstorm Days</th>
</tr>
</thead>
<tbody>
<tr>
<td>January</td>
<td>0</td>
</tr>
<tr>
<td>February</td>
<td>≤0.1</td>
</tr>
<tr>
<td>March</td>
<td>0.2</td>
</tr>
<tr>
<td>April</td>
<td>0.7</td>
</tr>
<tr>
<td>May</td>
<td>1.6</td>
</tr>
<tr>
<td>June</td>
<td>2.3</td>
</tr>
<tr>
<td>July</td>
<td>2.1</td>
</tr>
<tr>
<td>August</td>
<td>2.0</td>
</tr>
<tr>
<td>September</td>
<td>0.8</td>
</tr>
<tr>
<td>October</td>
<td>0.2</td>
</tr>
<tr>
<td>November</td>
<td>0</td>
</tr>
<tr>
<td>December</td>
<td>≤0.1</td>
</tr>
<tr>
<td>Year</td>
<td>9.9</td>
</tr>
</tbody>
</table>

Note: Information in this table is from PNL 11794 1998 Climatological Data Summary 1997 with Historical Data Pacific Northwest National Laboratory Richland Washington

Thunderstorms can theoretically occur during any month of the year, but none have been observed in November or January. Thunderstorm season is essentially from April through September.

Thunderstorms are classified by the National Weather Service as severe when wind gusts exceed 58 mi/h and/or hail size equals or exceeds 1 in in diameter. Only 1.9% of all thunderstorm events observed at the HMS have been severe storms (PNL-6415). All severe storms met the wind gust criteria, but hail was seldom observed at the HMS. The maximum recorded number of days with hail in a year is two.

Large differences in electrical potential between cloud and earth can occur during thunderstorms and can lead to lightning strikes. There is an average of 0.06 lightning strikes per year per square kilometer at the Hanford Site (WHC-SD-WM-ES-387).

14124 Extreme Winds and Tornadoes  Table 1-8 shows the maximum recorded peak gusts at the HMS for the period 1945 through 1994. The highest peak gust measured at the 15-m (50-ft) height was 80 mi/h in January 1972 (PNL-11794). Estimates of extreme winds, based on peak gusts observed from 1945 through 1980, are given in PNL-4622. The extreme peak gust at
50 ft for a return period of 100 years is estimated to be 86 mi/h. The return period on gusts of 114 km/h (70 mi/h) at 50 ft is 10 years.

Two probabilistic wind hazard assessments have been completed for the Hanford Site. The first assessment was completed by Lawrence Livermore National Laboratory and reported in UCRL-53526, *Natural Phenomena Hazards Modeling Project: Extreme Wind/Tornado Hazard Models for Department of Energy Sites*. This assessment was based on more than 30 years of pre-1979 Hanford Site wind data. The results are shown in Figure 1-12. The wind speeds are fastest mile and therefore, for an equivalent probability considerably lower than the peak gusts illustrated in Table 1-8. Further, the hazard curves are based on winds 30 ft above ground surface and the peak gusts in Table 1-8 were measured at 50 ft above ground surface. Wind speed increases with distance above the ground surface. A second study, NUREG/CR-4492, *Methodology for Estimating Extreme Winds for Probabilistic Risk Assessments*, describes a procedure for estimating extreme wind probabilities. The application of this methodology to Hanford Site data including post-1979 data resulted in hazard curves also shown in Figure 1-12.

### Table 1-8: Maximum Peak Gusts 50 Feet off the Ground at the Hanford Meteorological Station for the Period 1945 through 1997

<table>
<thead>
<tr>
<th>Month</th>
<th>Peak gust speed (mi/h)</th>
<th>Direction of peak</th>
<th>Year</th>
</tr>
</thead>
<tbody>
<tr>
<td>January</td>
<td>80</td>
<td>SW</td>
<td>1972</td>
</tr>
<tr>
<td>February</td>
<td>65</td>
<td>SW</td>
<td>1971</td>
</tr>
<tr>
<td>March</td>
<td>70</td>
<td>SW</td>
<td>1956</td>
</tr>
<tr>
<td>April</td>
<td>73</td>
<td>SSW</td>
<td>1972</td>
</tr>
<tr>
<td>May</td>
<td>71</td>
<td>SSW</td>
<td>1948</td>
</tr>
<tr>
<td>June</td>
<td>72</td>
<td>SW</td>
<td>1957</td>
</tr>
<tr>
<td>July</td>
<td>69</td>
<td>WSW</td>
<td>1979</td>
</tr>
<tr>
<td>August</td>
<td>66</td>
<td>SW</td>
<td>1961</td>
</tr>
<tr>
<td>September</td>
<td>65</td>
<td>SSW</td>
<td>1953</td>
</tr>
<tr>
<td>October</td>
<td>72</td>
<td>SW</td>
<td>1997</td>
</tr>
<tr>
<td>November</td>
<td>67</td>
<td>WSW</td>
<td>1993</td>
</tr>
<tr>
<td>December</td>
<td>71</td>
<td>SW</td>
<td>1955</td>
</tr>
<tr>
<td>Year</td>
<td>80</td>
<td>SW</td>
<td>January 1972</td>
</tr>
</tbody>
</table>

Note: Information in this table is from PNL 11794 1998 *Climatological Data Summary 1997 with Historical Data*. Pacific Northwest National Laboratory, Richland, Washington.
The wind hazard annual probability of exceedance for performance categories 1 and 2 is $2 \times 10^{-2}$ (DOE-STD-1020-94). On Figure 1-12, this is about 55 mi/h on the UCRL-53526 curve and about 58 mi/h on the NUREG/CR-4492 curve. However, a minimum design 3-second gust wind speed of 85 mi/h, which is equivalent to 70 mi/h, is required by the American Society of Civil Engineers in ASCE-7-95, *Minimum Design Loads for Building and Other Structures*, and recommended by DOE-STD-1020-94. Therefore, the Hanford Site design basis wind speed for performance categories 1 and 2 is 85 mi/h, 3-second wind gust or 70 mi/h fastest mile. Wind speeds are at 30 ft off the ground. Note that maximum (approximately 3-second) gusts in Table 1-8 are at 50 ft above the ground surface. Windspeed increases as the distance from the ground increases.

The straight wind hazard exceedance probability for performance category 3 is $1 \times 10^{-3}$. On Figure 1-12, this is approximately 67 mi/h on the UCRL-53526 curve and 72 mi/h on the NUREG/CR-4492 curve. In DOE-STD-1020-94, the minimum straight wind speed for a performance category 3 design is 80 mi/h (equivalent to a 95 mi/h, 3-second gust) which is higher than either of the wind hazard studies and is, therefore, the design basis for the Hanford Site.

Hanford Standard Architectural-Civil Design Criteria (DOE-RL-HPS-SDC-41) establishes the wind load requirements for the existing K Basins structures. Nonreactor safety class 1 structures, systems, components, and equipment shall be designed for wind loads using the following criteria:

- **Fastest-mile speed**: 90 mi/h at a height of 33 ft
- **Importance factor**: 1
- **Exposure class**: C
- **Missile (horizontal)**: 2 x 4 in timber plank, 15 lb, at 50 mi/h maximum trajectory height of 50 ft

The intersection of the straight-wind and tornado hazard curves determines whether tornadoes should be included in the design and evaluation criteria for DOE facilities. (UCRL-53526) If the exceedance probability at the intersection is less than $2 \times 10^{-4}$, straight winds control the design criteria. In Figure 1-12, this intersection is less than $2 \times 10^{-5}$ for both curves. Therefore, following DOE guidance, the Hanford Site does not have a DOE design basis tornado and no tornado is required for the K Basins analyses.

The NRC tornado criteria were followed at the CVDF and CSB. This was achieved by following the same requirements as the Washington Public Power Supply System. The NRC staff have proposed dividing the United States into two regions and making the total rotational and translational wind speeds for sites west of the Rocky Mountains 200 mi/h (SECY-93-087). The basis for this consideration is a tornado hazard study, NUREG/CR-4461 *Tornado Climatology of the Contiguous United States*. The Washington Public Power Supply System requested a revision of the tornado criteria for WNP-2 (Parrish 1995) based upon the NRC staff's proposal for...
tornado wind speeds west of the Rocky Mountains and on the staff's acceptance of the design basis tornado characteristics in NUREG-1503, *Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design*. The impact velocities for the proposed tornado missile spectrum based on the reduced wind speed of 200 m/h were calculated by the Washington Public Power Supply System. The NRC accepted the proposed revision (Clifford 1996). These same criteria have been applied to the CVDF and CSB.

- **Wind speed**
  - 200 m/h total
  - 160 m/h rotational
  - 40 m/h translational

- **Pressure drop**
  - 0.90 lb/in² at 0.30 lb/in²/s

For the CVDF and CSB, the need for protection of specific targets is addressed on a risk assessment basis. By this process, an estimate is made of the probability that a tornado-generated missile will strike a particular target and fail the target in such a way that unacceptable consequences result. For facilities with relative small targets areas such as the CVDF, the risk assessment can simply investigate the probability that a missile strikes a target. If the probability obtained of a missile striking and failing a target, or simply striking the target, satisfies established acceptance criteria, then no physical barriers against missiles need be provided.

The use of probabilistic risk assessment techniques to establish the need to provide tornado-generated missile protection for specific targets has been accepted by the NRC for several years and is well documented in their review and design guidance. For example, the NRC Standard Review Plan (NUREG-0800) Section 3.5.15 "Site Proximity Missiles (Except Aircraft)" includes the following statement relative to low risk of exposure to the public:

This requirement is met if the probability of site proximity missiles impacting the plant and causing radiological consequences greater than 10 CFR Part 100 exposure guidelines is less than about 10⁻⁷ per year (see SRP Section 2.2.3).

The NRC Standard Review Plan (NUREG-0800) Section 2.2.3 "Evaluation of Potential Accidents" accepts 10⁶ events/yr for offsite hazards exceeding the guidelines in Title 10 *Code of Federal Regulations* Part 100 "Reactor Site Criteria" (10 CFR 100), if when accompanied by reasonable qualitative arguments, the realistic value can be shown to be lower. These provisions have remained in the 1996 draft revision to Standard Review Plan 2.2.3. For the CSB, an acceptance criterion to not exceed the 5 rem (50 mSv) limit of Title 10 *Code of Federal Regulations* Part 72 "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" Section 72.106 (10 CFR 72), at the controlled area boundary should be used.
The probability of a tornado occurring near the site and generating a missile that results in acceptable consequences can be expressed as follows

$$P_{TM} = P_s \times P_{ms} \times P_f \times P_D$$

where

- $P_{TM}$ = The annual probability of a tornado missile striking such that an unacceptable radiological or chemical release occurs
- $P_s$ = The tornado annual point strike frequency (events/yr)
- $P_{ms}$ = The probability of a tornado-generated missile impacting a specific target (this includes the probabilities of the missile existing on the site being lifted and striking the target of a given area)
- $P_f$ = The probability of target failure as a result of being struck by a missile (this considers that not all missiles that strike the target will be of sufficient energy to fail the target)
- $P_D$ = The probability, given the failure of the target by a missile, that unacceptable radiological or chemical release occurs (this considers that not all target failures will be of sufficient severity to result in unacceptable consequences)

The above expression is simplified in that the analyses are usually performed for a range of tornado intensities and potential missiles. The tornado intensities considered are based upon available local severe weather data. The missiles considered vary in terms of their density on the site, ability to be lifted by a tornado of a given intensity, and ability to damage the target. The missiles appropriate for consideration are often established by a site survey with a radius of about 2000 ft.

It should be noted in the above equation that the missile selection is based upon realistic selection of potential missiles based upon a site survey. The results of the tornado missile risk assessment for the CVDF are provided in *Probabilistic Risk Analysis Tornado Missile Hazard to Cold Vacuum Drying System and Hot Conditioning System* (Beary et al 1996). The analysis concluded that the tornado missile strike frequency to exposed CVDF critical area and equipment is well below the NRC frequency criterion of $10^{-7}$ per year for a realistic analysis.

The results of the tornado missile risk assessment for the CSB are provided in Letter FRF-2855 *Tornado Loading Design Criteria - Revised* (Jacobs 1996). This assessment has been accepted by DOE (Sellers 1996). The estimated frequency for a missile strike on the CSB is $4 \times 10^9$ events/yr, which is well below the NRC criterion of $1 \times 10^6$ events/yr. The ability of the CSB to withstand the tornado loadings is addressed in Section A4.3 in Annex A.
Dust and Blowing Dust  Dust and blowing dust are weather phenomena that occur with some frequency at the Hanford Site. Dust and blowing dust are recorded at the HMS when horizontal visibility is reduced to 6 mi or less. Dust is carried into the area from distant sources and may or may not occur during strong winds. Dust has been observed with wind speeds ranging from 4 mi/h to 30 mi/h. Blowing dust occurs with strong winds when dust is being picked up locally. While both dust and blowing dust occur at the HMS, blowing dust is the most commonly observed.

Table 1-9 shows the average number of days of recorded dust or blowing dust for the period 1945 through 1997. The average number of days per year with dust or blowing dust is five. The greatest number of such days in any year is 20, while the fewest is 0. The greatest number of days with dust or blowing dust in any month was nine in May 1980. Dust and blowing dust occur most frequently between March and May and again in September and occur least frequently during November and December.

Table 1-9  Average Number of Days of Dust or Blowing Dust at the Hanford Meteorological Station for the Period 1945 through 1997

<table>
<thead>
<tr>
<th>Month</th>
<th>Days of dust or blowing dust</th>
</tr>
</thead>
<tbody>
<tr>
<td>January</td>
<td>0.4</td>
</tr>
<tr>
<td>February</td>
<td>0.4</td>
</tr>
<tr>
<td>March</td>
<td>0.5</td>
</tr>
<tr>
<td>April</td>
<td>0.6</td>
</tr>
<tr>
<td>May</td>
<td>0.7</td>
</tr>
<tr>
<td>June</td>
<td>0.4</td>
</tr>
<tr>
<td>July</td>
<td>0.4</td>
</tr>
<tr>
<td>August</td>
<td>0.3</td>
</tr>
<tr>
<td>September</td>
<td>0.5</td>
</tr>
<tr>
<td>October</td>
<td>0.3</td>
</tr>
<tr>
<td>November</td>
<td>0.2</td>
</tr>
<tr>
<td>December</td>
<td>0.2</td>
</tr>
<tr>
<td>Year</td>
<td>4.7</td>
</tr>
</tbody>
</table>

Note: Information in this table is from PNL 11794 1998 Climatological Data Summary 1997 with Historical Data Pacific Northwest National Laboratory Richland, Washington.
14126 Fog and Sky Cover Table 1-10 shows the average monthly and annual number of days with fog and dense fog for the period 1945 through 1997. Fog is reported any time horizontal visibility is reduced to 6 mi or less because of suspension of water droplets in the surface layer of the atmosphere. Dense fog is reported when horizontal visibility is reduced to 0.25 mi or less. Most of the fog at the HIMS is radiation fog, a common type of fog that forms on nights characterized by light wind, clear sky, and moist air in the lower levels of the atmosphere. Nearly 90% of both fog and dense fog occur during the late autumn and winter months. The longest duration for fog on the Hanford Site was 114 hours in December 1985 and the longest duration for dense fog was 47 hours in December 1957.

<table>
<thead>
<tr>
<th>Month</th>
<th>Days with fog</th>
<th>Days with dense fog</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Average</td>
<td>Maximum</td>
</tr>
<tr>
<td>January</td>
<td>11 2</td>
<td>25</td>
</tr>
<tr>
<td>February</td>
<td>6 7</td>
<td>20</td>
</tr>
<tr>
<td>March</td>
<td>2 1</td>
<td>10</td>
</tr>
<tr>
<td>April</td>
<td>0 5</td>
<td>3</td>
</tr>
<tr>
<td>May</td>
<td>0 2</td>
<td>3</td>
</tr>
<tr>
<td>June</td>
<td>0 1</td>
<td>2</td>
</tr>
<tr>
<td>July</td>
<td>&lt;0 1</td>
<td>1</td>
</tr>
<tr>
<td>August</td>
<td>0 1</td>
<td>1</td>
</tr>
<tr>
<td>September</td>
<td>0 3</td>
<td>2</td>
</tr>
<tr>
<td>October</td>
<td>2 0</td>
<td>9</td>
</tr>
<tr>
<td>November</td>
<td>9 8</td>
<td>19</td>
</tr>
<tr>
<td>December</td>
<td>14 2</td>
<td>25</td>
</tr>
<tr>
<td>Year</td>
<td>47 4</td>
<td>84</td>
</tr>
</tbody>
</table>

Note: Information in this table is from PNL 11794 1998 Climatological Data Summary 1997 with Historical Data Pacific Northwest National Laboratory Richland Washington.

Horizontal visibility is reduced to 6 mi or less.

*Horizontal visibility is reduced to 0.25 mi or less.
The term sky cover is used to express the portion of the celestial dome that is (1) covered but not necessarily hidden by clouds or obscuring phenomena aloft, (2) hidden by an obscuring phenomenon on the ground (e.g., fog, smoke), or (3) a combination of (1) and (2). Sky cover is then determined hourly by scanning the sky and estimating the number of tenths that are covered (zero denotes clear sky and ten denotes overcast). Each day can be assigned a designation depending upon the average sky cover from sunrise to sunset. The designations are clear (zero to three tenths), partly cloudy (four to seven tenths), and cloudy (eight to ten tenths). Table 1-11 shows the average monthly sunrise to sunset sky values from 1946 through 1997 and the number of days of clear, partly cloudy, and cloudy days for the years 1954 to 1997. During the period of record (1954 through 1997), an average of 199 sunny days (sum of clear and partly cloudy days) were recorded per year at the HMS.

Table 1-11  Average Sky Cover (Sunrise to Sunset) for the Period 1946 through 1997 and Number of Clear, Partly Cloudy, and Cloudy Days for the Period 1954 through 1997 at the Hanford Meteorological Station

<table>
<thead>
<tr>
<th>Month</th>
<th>Average sky cover (in tenths)</th>
<th>Number of clear days</th>
<th>Number of partly cloudy days</th>
<th>Number of cloudy days</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Average</td>
<td>Maximum</td>
<td>Minimum</td>
<td>Average</td>
</tr>
<tr>
<td>January</td>
<td>8.0</td>
<td>9.2</td>
<td>4.3</td>
<td>3</td>
</tr>
<tr>
<td>February</td>
<td>7.5</td>
<td>9.3</td>
<td>5.9</td>
<td>4</td>
</tr>
<tr>
<td>March</td>
<td>6.7</td>
<td>8.5</td>
<td>4.9</td>
<td>6</td>
</tr>
<tr>
<td>April</td>
<td>6.5</td>
<td>8.1</td>
<td>3.7</td>
<td>6</td>
</tr>
<tr>
<td>May</td>
<td>5.9</td>
<td>8.1</td>
<td>3.6</td>
<td>8</td>
</tr>
<tr>
<td>June</td>
<td>5.2</td>
<td>7.0</td>
<td>2.8</td>
<td>10</td>
</tr>
<tr>
<td>July</td>
<td>3.1</td>
<td>5.0</td>
<td>0.9</td>
<td>19</td>
</tr>
<tr>
<td>August</td>
<td>3.3</td>
<td>5.9</td>
<td>0.6</td>
<td>18</td>
</tr>
<tr>
<td>September</td>
<td>3.9</td>
<td>6.7</td>
<td>1.4</td>
<td>15</td>
</tr>
<tr>
<td>October</td>
<td>5.7</td>
<td>8.0</td>
<td>3.3</td>
<td>10</td>
</tr>
<tr>
<td>November</td>
<td>7.5</td>
<td>9.1</td>
<td>5.2</td>
<td>5</td>
</tr>
<tr>
<td>December</td>
<td>8.0</td>
<td>9.3</td>
<td>6.4</td>
<td>4</td>
</tr>
<tr>
<td>Year</td>
<td>5.9</td>
<td>6.6</td>
<td>5.1</td>
<td>110</td>
</tr>
</tbody>
</table>

Note: Information in this table is from PNL 11794, 1998 Climatological Data Summary 1997 with Historical Data, Pacific Northwest National Laboratory, Richland, Washington.
14127 Relative Humidity  The annual mean relative humidity recorded at the HMS is 54%, with the highest average monthly relative humidity (80%) occurring in December and the lowest average monthly relative humidity (33%) occurring in July. Daily relative humidity can change 20% to 30% between early morning and late afternoon, except for the winter months when changes are less pronounced (PNL-4622). Higher relative humidity can be expected at locations near the Columbia River and at some locations on the Site periphery where there is increased airborne water vapor from the Columbia River, Yakima River, and irrigated land.

14128 Dispersion Climatology  Atmospheric dispersion is a function of wind speed, duration and direction of wind, atmospheric stability, and mixing depth. Dispersion conditions are generally favorable if winds are moderate to strong, if the atmosphere is of neutral or unstable stratification, and if there is a deep mixing layer. These conditions exist more than half the time at the Hanford Site in the summer. A less favorable dispersion condition may occur when the wind speed is light and the mixing layer is shallow. This condition is most common in the winter when moderately to extremely stable stratification exists about two-thirds of the time (PNL-6415). Less favorable dispersion conditions also occur periodically for surface and low-level releases in all seasons from about sunset to about an hour after sunrise as a result of ground-based temperature inversions and shallow mixing layers.

Mixing layers at the HMS are estimated remotely using a Doppler acoustic sounder. Variations in the mixing layer are summarized in Table 1-12. The shallowest mixing layers are usually found in the nighttime during the winter months. Mixing layers are usually larger in the daytime during the summer months.

Table 1-12  Percent Frequency of Occurrence of Mixing Layer Thickness by Season and Time of Day at the Hanford Site

<table>
<thead>
<tr>
<th>Mixing layer (m)</th>
<th>Winter</th>
<th></th>
<th>Summer</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Night</td>
<td>Day</td>
<td>Night</td>
</tr>
<tr>
<td>&lt;250</td>
<td>65 7</td>
<td>35 0</td>
<td>48 5</td>
</tr>
<tr>
<td>250-500</td>
<td>24 7</td>
<td>39 8</td>
<td>37 1</td>
</tr>
<tr>
<td>&gt;500</td>
<td>9 6</td>
<td>25 2</td>
<td>14 4</td>
</tr>
</tbody>
</table>


Low-level inversions can also be associated with stagnant air in stationary high-pressure systems that occur primarily in the winter months. The probability of extended periods of low-level inversions is estimated in BNWL-1605, Climatology of the Hanford Area. The probability of an inversion period extending more than 12 hours varies from a low of about 10%.
in May and June to a high of about 64% in September and October. These probabilities decrease rapidly for durations of greater than 12 hours. Table 1-13 summarizes the probabilities associated with extended surface-based inversions.

**Table 1-13 Percent Probabilities for Extended Periods of Surface-Based Inversions at the Hanford Site**

<table>
<thead>
<tr>
<th>Months</th>
<th>Inversion duration</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>12 hours</td>
</tr>
<tr>
<td>January-February</td>
<td>54.0</td>
</tr>
<tr>
<td>March-April</td>
<td>50.0</td>
</tr>
<tr>
<td>May-June</td>
<td>10.0</td>
</tr>
<tr>
<td>July-August</td>
<td>18.0</td>
</tr>
<tr>
<td>September-October</td>
<td>64.0</td>
</tr>
<tr>
<td>November-December</td>
<td>50.0</td>
</tr>
</tbody>
</table>

Note: Information in this table is from PNL 6415 1995 *Hanford Site National Environmental Policy Act (NEPA) Characterization Rev 7 Pacific Northwest Laboratory Richland Washington*

The joint frequency distribution of hourly averaged wind data from the 100 Area meteorological towers for the 9-year period January 1983 to December 1991 are provided in PNL-3777, *Recommended Environmental Dose Calculation Methods and Hanford-Specific Parameters*. The data are given for various wind speed classes and stability classes. Stability class was determined using the method based on Pasquill (1961) which is a parameterization scheme based on time of year, time of day, wind speed, cloud cover, and visibility. Cloud cover and visibility were measured at the HMS but should be representative of the 100 Area.

The percent frequency of occurrence of each of the stability classes for the 9-year period for the 100 Area 10 m tower is given in Table 1-14. Stability Class D, relatively neutral dispersion conditions occurred 28% of the time. Conditions favorable for turbulent dispersion represented by the unstable classes A, B, and C occurred about 23% of the time. The stable classes E, F, and G indicative of conditions unfavorable for turbulent dispersion occurred about 49% of the time.

To support the accident analyses in the facility FSAR Annexes, air transport factors (X/Q and X/Q') were calculated (HNF-SD-SNF-TI-059). These air transport factors represent the dilution of a contaminant by atmospheric turbulence and diffusion as the contaminant travels downwind. The symbol X/Q is the ratio of the average air concentration at the receptor to the average release rate at the release point. It is used to assess potential radiological dose and noncorrosive chemical concentration at downwind locations. The symbol X/Q' is the normalized...
peak air concentration at the center of a puff divided by the quantity released and is used to assess the consequences to a receptor for corrosive chemicals.

<table>
<thead>
<tr>
<th>Class</th>
<th>Percent occurrence per year</th>
</tr>
</thead>
<tbody>
<tr>
<td>A (extremely unstable)</td>
<td>13.9%</td>
</tr>
<tr>
<td>B (moderately unstable)</td>
<td>4.5%</td>
</tr>
<tr>
<td>C (slightly unstable)</td>
<td>4.3%</td>
</tr>
<tr>
<td>D (neutral)</td>
<td>28.3%</td>
</tr>
<tr>
<td>E (slightly stable)</td>
<td>26.3%</td>
</tr>
<tr>
<td>F (moderately stable)</td>
<td>15.7%</td>
</tr>
<tr>
<td>G (extremely stable)</td>
<td>6.8%</td>
</tr>
</tbody>
</table>

The locations of maximum exposure to the public and to onsite workers were determined by calculating the air transport factors for the 16 directional sectors at the distances representing the nearest location of the potential receptors. Consequences to collocated workers are assessed at 100 m from the facility and for K Basins and CVDF at the 100 Area Fire Station. For ground-level releases, the direction of maximum exposure for the 100 m location was found to be east of the facility. For elevated releases, the maximum exposure to the onsite receptor occurs at distances greater than 100 m. The location of the maximum onsite dose for elevated releases was determined by calculating the transport factors for various distances from 100 m to the Site boundary in all sectors.

Exposure to public receptors is assessed both at the current Site boundary and the near bank of the Columbia River. The location of maximum exposure at the current Site boundary was found to be 12.0 km west of the 100 K Area facilities for both ground-level and elevated releases. The location of maximum exposure at the river is 700 m west of the facility for ground-level releases, and 580 m northwest for releases of 0.5-hour to 2-hour duration.

The joint frequency distribution of hourly averaged wind data from the 200 East Area meteorological tower for the 9-year period January 1983 to December 1991 are provided in PNL-3777. Stability class was determined using the method based on Pasquill (1961), which is a parameterization scheme based on time of year, time of day, wind speed, cloud cover, and visibility. Cloud cover and visibility were measured at the HMS but should be representative of the 200 East Area. The percent frequency of occurrence of each of the stability classes for the 9-year period for the CSB is given in Table 1-1. Stability Class D, relatively neutral dispersion conditions, occurred 28% of the time. Conditions favorable for turbulent dispersion, represented...
by the unstable classes A, B, and C occurred about 23% of the time. The stable classes E, F, and G, indicative of conditions unfavorable for turbulent dispersion, occurred about 49% of the time.

<table>
<thead>
<tr>
<th>Class</th>
<th>Percent occurrence per year</th>
</tr>
</thead>
<tbody>
<tr>
<td>A (extremely unstable)</td>
<td>14%</td>
</tr>
<tr>
<td>B (moderately unstable)</td>
<td>46%</td>
</tr>
<tr>
<td>C (slightly unstable)</td>
<td>44%</td>
</tr>
<tr>
<td>D (neutral)</td>
<td>28%</td>
</tr>
<tr>
<td>E (slightly stable)</td>
<td>26.2%</td>
</tr>
<tr>
<td>F (moderately stable)</td>
<td>15.8%</td>
</tr>
<tr>
<td>G (extremely stable)</td>
<td>7.0%</td>
</tr>
</tbody>
</table>

Additional facility-specific meteorological information is provided in the facility FSAR Annexes.

14.2 Hydrology

The Hanford Site is located within the Columbia River drainage basin with the 100 K Area situated along the river and the 200 East Area located in the 200 Area plateau. The Columbia River is the principal hydrologic surface feature in the area. Other surface hydrologic features in the region include the Yakima River, Snake River, and the Walla Walla River. However, none of these other rivers have impact on the 100 K Area or the 200 East Area.

The Hanford Site is underlain by unconsolidated to semiconsolidated sediments of the Hanford and Ringold Formations. The Columbia River Basalt Group (CRBG) and associated sedimentary interbeds lie beneath the sediments. The geology of the Hanford Site is described in Section 14.3.3.

The vadose zone at the Hanford Site is comprised mainly of unconsolidated gravels and sands. Its thickness ranges from 0 ft at the river bank and West Lake to over 328 ft in the 200 Areas. The vadose zone at the K Basins and CVDF site is approximately 70 ft thick.

The saturated sediments of the Hanford and Ringold Formations make up a series of aquifers and aquitards. In general, sand- and gravel-dominated stratigraphic units form aquifers, and fine-grained deposits form aquitards. The shallowest suprabasalt aquifer is unconfined.
beneath most of the Hanford Site. Confined aquifers are present in the sedimentary interbeds and/or interflow zones between dense basalt flows of the CRBG. The main water-bearing portions of the interflow zones are networks of interconnecting vesicles and fractures of the flow tops and flow bottoms.

The remainder of this section describes the surface water features, vadose zone, and aquifers as they relate to the 100 K and 200 East Areas of the Hanford Site.

14.2.1 Surface Water

This section describes the surface water bodies and the river flooding potential at the CVDF and K Basins sites from the Columbia River and for the potential at the CSB site from the Columbia River, Yakima River, and Cold Creek. An overview of artificial surface water bodies in and near the 200 Areas is also presented.

14.2.1.1 Water Bodies and Flooding Potential at the Cold Vacuum Drying Facility and K Basins

West Lake

West Lake is the only natural lake on the Hanford Site (Figure 1-13).

Columbia River

The Columbia River flows through the Hanford Site and forms part of its eastern boundary (Figure 1-13). The section of the river flowing through the Site, known as the Hanford Reach, is free-flowing and extends from Priest Rapids Dam, upstream of the Site, to the 300 Area. Flow through the Hanford Reach fluctuates significantly and is controlled by Priest Rapids Dam. Discharge ranges from about 36,000 ft³/s to nearly 400,000 ft³/s. The average flow of the Columbia River near the Priest Rapids Dam is about 120,000 ft³/s. Currently, the minimum discharge at points along the river is regulated by law. Minimum allowable discharges above the Hanford Site are calculated to be sufficient to not impact the functioning of raw water intake structures which supply much of Hanford's process and potable water.

The Columbia River originates in the mountains of eastern British Columbia, Canada, and drains an area of approximately 27,300 mi² en route to the Pacific Ocean (Figure 1-14). Flow on the Columbia River is regulated by seven upstream dams within the United States and several in Canada. The three dams with the largest reservoirs upstream from the Hanford Site are the Mica Dam and Hugh Keenleyside Dam in Canada and the Grand Coulee Dam in the United States. The dam-controlled flow of the Columbia River results in a lower flood hazard for high-probability floods (e.g., 100-year flood), but dam-failure scenarios are significant contributors for very-low-probability floods. A flood scenario involving a 50% breach of Grand Coulee Dam results in a flood level of about 480 ft above mean sea level at the 100 K Area (RLO-76-4).

The design basis flood level for the K Basins is identified by HPS-SDC-4.1 Standard Architectural-Civil Design Criteria Design Loads for Facilities, for safety class 1 structures and components (SSCs) (performance class 3) at 100-K Area as 432 ft, and there are no flood requirements for safety class 2, or 3 SSCs (performance class 1 and 2). The 181-KE River Pump Station is located at an elevation of 421 ft mean sea level, the 105-K fuel storage basins are located at an elevation of 465 ft mean sea level and the support equipment located in the 165-KE, 183-KE, and 190-KE are at an elevation of 459 ft mean sea level or higher.
A probabilistic flood hazard assessment of the Columbia River was performed as part of a safety evaluation of the Hanford Site N Reactor and documented in UCRL-21069, *Probabilistic Flood Hazard Assessment of the Department of Energy N-Reactor Hanford, Washington*. The most extreme potential flood scenarios result from dam failure on the Columbia River. Specifically, failures of Grand Coulee Dam and Mica Dam pose the greatest threat. The results of the flood hazard assessment show that the greatest contribution to the likelihood of flooding is dam failure initiated by seismic events. Conservative estimates of the seismic capacity of upstream dams were made primarily on engineering judgment. Further, the likelihood of random dam failure was based primarily on engineering judgment and the historic frequency of dam failures. See UCRL-21069 for a detailed discussion of the hazard assessment.

The performance category 3 flood level (10^-4 annual probability of exceedance [UCRL-21069]) at the 100 K Area is approximately 450 ft above mean sea level (WHC-SD-SNF-DB-010). The Columbia River flood design for the WNP-2 is 424 ft above median sea level (NUREG-0892) which corresponds to a 25% breach of the Grand Coulee Dam. The elevation of this flood is about 460 ft above mean sea level at the 100 K Area. This is the flood level to be applied to performance category 3 aspects at the CVDF. As discussed earlier in this section, the design flood level for the K Basins is 432 ft above mean sea level.

Extreme flooding on the Columbia River can also occur during extreme meteorologic and hydrologic events. Historically, the largest floods have occurred when rapid snow melt was accompanied by spring rain. Such floods are controlled by the dam system and are predicted to be less severe than comparable pre-dam record floods.

**Yakima River** The Yakima River forms part of the southern boundary of the Hanford Site (Figure 1-13). The Yakima River and its tributary, Cold Creek, are not a flood hazard for the 100 K Area.

14212 Water Bodies and Flooding Potential at the Canister Storage Building Site
A flood scenario involving a 50% breach of Grand Coulee Dam (RLO-76-4) results in a flood level of about 470 ft above mean sea level at Columbia River kilometer mile 365 (Figure A1-1 in Annex A), the closest flood route to the 200 Plateau. This flood level is about 230 ft below the ground surface at the CSB. The CSB is a flood-dry site with respect to the Columbia River (ANSI/ANS-2 8-1992).

The performance category 3 flood level (10^-4 annual probability of exceedance [UCRL-21069]) at river mile 365, the closest flood route to the 200 Area Plateau and the CSB (Figure A1-4 in Annex A), is approximately 435 ft above mean sea level, which is lower than the 50% breach scenario (WHC-SD-SNF-DB-009). The 200 Areas are dry sites with respect to Columbia River flooding and such flooding need not be considered in design or accident scenarios.

**Yakima River** The Yakima River forms part of the southern boundary of the Hanford Site (Figure 1-13). The Yakima River is approximately 12 mi south and more than 200 ft below the 200 Areas. The Yakima River is not a flood hazard for the CSB site.

fsar01 sur 1-31 November 1999
Cold Creek, a tributary of the Yakima River, and its tributary, Dry Creek are ephemeral streams within the Yakima River drainage basin (Figure 1-15). The Cold Creek watershed located in the southwestern portion of the Hanford Site, extends about 10 mi up the Cold Creek and Dry Creek valleys. A flood risk analysis of Cold Creek was conducted in 1980 (RHO-BWI-C-120) to determine the probable maximum flood for the Cold Creek system (Figure 1-15). The recurrence interval is not estimated for this flood, however each occurrence would be a flash flood of short duration. The probable maximum flood reaches an elevation of about 640 ft on the southwestern portion of 200 West Area and is not a hazard for the CSB.

14213 Artificial Surface Water Bodies In the past, numerous artificial surface water bodies were present in the 200 Areas. Effluent disposal sites (e.g., cribs, ponds, ditches) allowed waste water to infiltrate the ground and, in many cases, to affect groundwater flow and chemistry. Today only B Pond and the Treated Effluent Disposal Facility, both located east of 200 East Area receive significant volumes of effluent.

1422 Vadose Zone The term "vadose zone" is used here to refer to unsaturated sediments between the water table and the ground surface. Flow of water through the vadose zone is a function of the moisture content, matric potential, and unsaturated hydraulic conductivity for each hydrostratigraphic unit. In general, water flows and spreads laterally at a much greater rate in fine-grained units than in coarse-grained units. Fine-grained units in the vadose zone significantly influence the lateral distribution of water and the flux of water to the uppermost aquifer. Coarse-grained units may impede the flux of water through the vadose zone because of the formation of a capillary pressure barrier between the coarse-grained units and overlying fine-grained units.

The vadose zone beneath the 100 K Area includes backfill in areas near the buildings or facilities, Holocene surficial deposits, the Hanford formation, and the uppermost part of the Ringold Formation (unit E). The vadose zone varies from being absent along the shore of the Columbia River to over 150 ft in the southeastern portion of the 100 K Area. At the CVDF the vadose zone is approximately 75 ft thick. Most of the vadose zone lies within the gravel-dominated, relatively permeable facies of the Hanford formation and the upper part of Ringold Formation unit E (predominantly a fluvial gravel).

The vadose zone in the 200 East Area is comprised of interlayered gravel, sand, silt, and "mud" (i.e., silt and clay). In some locations the basalt has been uplifted and composes part of the vadose zone. Thickness of the vadose zone in the 200 East Area ranges from 121 ft near B Pond to 341 ft near the southern border of the area (DOE/RL-93-88). Fine-grained units in the Hanford formation and the Ringold Lower Mud unit significantly influence the lateral distribution and flux of water in the 200 East Area.

The fine-grained sedimentary units in the 200 East Area may produce perched water conditions near active waste disposal facilities. Perched water results when a discontinuous impermeable layer in the vadose zone traps discontinuous water above the saturated zone. Perched water has been observed to the east of the 241-BY tank farm and above the Lower Mud near B Pond (Figure 1-13) and the Treated Effluent Disposal Facility.
Routson and Johnson (1990) estimated the magnitude of recharge from natural precipitation on the 200 Areas plateau. Two sources of data were used in their evaluation: (1) moisture accumulated for 13 years in an 60-ft-deep 200 East Area lysimeter, and (2) distribution of $^{137}$Cs in a 200 West Area solid-waste burial ground in which contaminated soil had been buried for 10 years. There had been no detectable moisture accumulation in the bottom of the lysimeter from 1972 to 1985. Thus, the recharge rate was approximately zero over the 13-year period. Cesium-137 contaminated soil was placed in a burial ground in 1970. After 10 years the distribution of $^{137}$Cs was determined from core-drilled samples. No $^{137}$Cs above background was detected beneath the bottom of the trench, indicating there was no measurable downward movement, however, $^{137}$Cs was detected above the trench, most likely from evapotranspiration.

Routson and Johnson (1990) conclude that there appears to be very little potential for downward movement of contaminants in the upper vadose zone beneath much of the 200 Areas plateau under natural or minimally disturbed conditions. Highly disturbed sites with little vegetation (e.g., the CSB site) may allow more movement of water through the vadose zone.

PNL-10285, *Estimated Recharge Rates at the Hanford Site* compiled estimates of natural recharge on the Hanford Site. Recharge through the vadose zone was 3.4 in/yr in a gravel-covered, unvegetated lysimeter constructed to mimic the surface conditions at the tank farms.

Site-specific variables must be considered to estimate time for water to travel through the vadose zone. Variables include the amount of moisture infiltrating, the grain size and stratigraphy of the sediments, the amount of moisture already in each sedimentary layer, and the hydraulic conductivity. WHC-SD-EN-TI-014, *Hydrogeologic Model for 200 West Groundwater Aggregate Area* contains calculated travel times in four examples for the 200 West Area with steady recharge rates of 0.2 and 2.0 in/yr. Calculated travel times range from 150 years to over 2,000 years. The methods applied in WHC-SD-EN-TI-014 are not valid for transient conditions (e.g., spills, pipe leaks).

1423 Aquifers Two major aquifer systems lie beneath the 100 K Area and the 200 Areas: the suprabasalt aquifer system and the basalt and interbed aquifer system (see Figure 1-16). Stratigraphy of the 100 K Area is discussed in greater detail in Section 143. See Section A143 of Annex A for the stratigraphy of the 200 East Area.

14231 Suprabasalt Aquifer System

100 K Area The suprabasalt aquifer (Figure 1-16) in the vicinity of the 100 K Area is within the Ringold Formation and includes a series of confining and water-producing zones (WHC-SD-ER-TI-003). The upper part of the aquifer is an unconfined zone of generally higher hydraulic conductivity within a predominantly gravel unit of the Ringold Formation. Below the gravel unit there are zones of lower hydraulic conductivity (fine-grained sediments) that alternate with layers of higher conductivity sand and gravel forming locally confined aquifers.
Water level data from wells in the unconfined zone of the upper aquifer system indicate that the water table is about 70 to 75 ft below the ground surface in the vicinity of the 105-KW and 105-KE Buildings and slopes gently toward the Columbia River. Throughout most of the year, when the Columbia River is in its lower stages, the water table slopes almost directly toward the Columbia River in a northwesterly direction. The level of the water table where groundwater enters the 100 K Area (southeast boundary near wells K-35 and K-36) is relatively constant at about 394.8 to 396.0 ft elevation (Figure 1-17).

When the Columbia River is in a sustained higher water level stage, groundwater flow in the immediate vicinity of the 100 K area is affected in three ways:

- The level of the water table near the river is raised
- The overall gradient of the water table within the 100 K Area decreases from approximately 0.005 to approximately 0.003
- The slope (line perpendicular to water table contours) of the water table is slightly more northerly than during lower stages of the Columbia River with a corresponding shift in flow direction to a slightly more northerly direction.

Additional suprabasalt groundwater information is presented in WHC-SD-EN-TI-280, *Groundwater Monitoring Results for the 100 K Area Fuel Storage Basins March to December 1994*.

**200 East Area**

The suprabasalt aquifer system under the 200 Areas occurs primarily within the sediments of the Hanford and Ringold Formations. These sediments contain interlayered coarse- and fine-grained units forming a series of aquifers and aquitards. The depth to the water table under the 200 East Area ranges from less than 131 ft near B Pond to approximately 341 ft west of the 200 East Area. The CSB site is at about 708 ft above mean sea level and the elevation of the bottom of the basemat is 659 ft above mean sea level. The water table at 413 ft above mean sea level is 246 ft below the CSB basemat.

The suprabasalt aquifer system in the 200 East Area includes the Ringold Formation and parts of the Hanford formation. Sandy gravels dominate the saturated Hanford formation and gravels dominate the Ringold Formation. Significant silt- and clay-dominated intervals are absent except in the southwestern part of the 200 East Area and east of B Pond where the Lower Mud sequence is found. The suprabasalt aquifer system ranges in thickness from zero where basalt is present above the water table to 197 ft in the south and west portions of the 200 East Area. At the CSB site, the suprabasalt aquifer is about 300 ft below surface and about 100 ft thick.

Horizontal hydraulic conductivity of units within the suprabasalt aquifer system varies considerably (Table 1-16). WHC-SD-EN-TI-019, *Hydrogeologic Model for the 200 East Groundwater Aggregate Area*, documents the compiled and mapped hydraulic conductivity data for the upper portion of the uppermost aquifer in the 200 West and 200 East Areas. The highest conductivity occurs in a channel stretching from northwest to southeast across the 200 East Area.
Table 1-16  Aquifer Properties in the 200 Areas

<table>
<thead>
<tr>
<th>Location</th>
<th>Interval</th>
<th>Hydraulic conductivity (m/d)</th>
<th>Transmissivity (m²/d)</th>
<th>Data source</th>
</tr>
</thead>
<tbody>
<tr>
<td>200 Areas (undifferentiated)</td>
<td>Elephant Mountain</td>
<td>0.07 to 569</td>
<td>RHO-ST-42*</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Member interflow zone</td>
<td></td>
<td></td>
<td>RHO-RE-ST-12P*</td>
</tr>
<tr>
<td></td>
<td>Rattlesnake Ridge</td>
<td>0.74 to 108</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>interbed</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>200 West</td>
<td>Ringgold unconfined aquifer</td>
<td>0.04 to 61</td>
<td>--</td>
<td>PNL-6820e</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.093 to 475</td>
<td></td>
<td></td>
</tr>
<tr>
<td>200 East</td>
<td>Top of unconfined aquifer</td>
<td>8 to 7,600</td>
<td>--</td>
<td>WHC-SD-EN-TI019d</td>
</tr>
<tr>
<td></td>
<td>(Hanford and Ringold)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>1.6 x 10E-05</td>
<td></td>
<td>PNL-6820e</td>
</tr>
</tbody>
</table>


Groundwater flow beneath the 200 East Area is complex because of the convergence of flow from the west (natural groundwater flow system) and from the east (artificial recharge from B Pond and the Treated Effluent Disposal Facility) (Figure 1-12). The high transmissivity beneath most of the 200 East Area causes hydraulic gradients to be very small. It is often difficult to define groundwater flow directions from water table maps of the 200 East Area, but contaminant plume maps of the 200 Areas indicate long-term trends in groundwater flow directions (PNL-10698). These plume maps indicate a north to northwest direction of flow in the extreme north central portion of the 200 East Area and a south to southeast direction of flow in the southeast portion of the 200 East Area.

Groundwater in the suprabasalt aquifer system is contaminated with various hazardous and radioactive constituents from waste previously discharged to the ground in the 200 Areas. The most significant contaminants include nitrate, carbon tetrachloride, tritium, and 99Tc. Contaminant plume maps for 1994 are presented in PNL-10698 *Hanford Site Ground-Water Monitoring for 1994*.

1 4 2 3 2 Basalt and Interbed Aquifer System The CRBG and associated sedimentary interbeds form a series of aquifers and aquitards. Generally, the dense basalt flows form aquitards and interflow zones (fractured basalt) and the sedimentary interbeds form aquifers (Figure 1-16).
The uppermost regionally extensive confined aquifer beneath the 200 Areas, K Basins and CVDF site is the Rattlesnake Ridge interbed and adjacent interflow zones

The interbed is 49 to 82 ft thick beneath the 200 Areas and generally thickens toward the west (RHO-ST-42 RHO-RE-ST-12P) Recharge to the Rattlesnake Ridge interbed aquifer occurs in the higher elevations to the west, north, and northeast. The flow of groundwater is generally toward the northeast beneath the 200 West Area and west to west-northwest beneath the 200 East Area (WHC-SD-ER-TI-003)

Beneath the 200 Areas the Rattlesnake Ridge interbed aquifer is generally separated from the unconfined aquifer system by the Elephant Mountain Member of the CRBG The Elephant Mountain Member is up to 115 ft thick. North of the 200 East Area the Elephant Mountain Member has been locally removed by erosion, allowing hydraulic communication between the Rattlesnake Ridge interbed aquifer and the suprabasalt aquifer system. RHO-RE-ST-12P, An Assessment of Aquifer Intercommunication with B Pond — Gable Mountain Pond Area of the Hanford Site determined that contamination found in the Rattlesnake Ridge interbed aquifer resulted from intercommunication through erosional windows between it and the overlyng unconfined aquifer. Transmissivity of the Rattlesnake Ridge interbed and the Elephant Mountain interflow zone are included in Table 1-16.

1 4 3 Geology

The Hanford Site lies within the Pasco Basin and the Pasco Basin lies within the Columbia Plateau a broad plain constructed from the Miocene CRBG and situated between the Cascade Range to the west and the Rocky Mountains to the east (Figure 1-18) In the central and western parts of the Columbia Plateau where the Hanford Site is located, the basalt is underlain predominantly by Tertiary continental sedimentary rocks and overlain by late Tertiary and Quaternary fluvial and glaciofluvial deposits. All these deposits were folded and faulted during the Cenozoic to form the present landscape of the region.

1 4 3 1 Physiographic Setting of the Hanford Site The physiography of the Columbia Basin, a subprovince of the Columbia Plateau is controlled by the late Cenozoic faulting and folding of the CRBG and the overlying sediments of the Ringold Formation (Figure 1-19) Surface topography has been modified within the past several million years by geomorphic processes related to Pliocene cataclysmic flooding. Holocene eolian activity, landsliding, and the Columbia River. The physiography of the Hanford Site is dominated by the low-relief plains northeast of the Site and by anticlinal ridges of the Yakima Folds physiographic region (Figure 1-19).

Cataclysmic flooding of the Hanford Site occurred when ice dams in western Montana and northern Idaho were breached allowing large volumes of water to spill across eastern and central Washington. The last major flood occurred about 13,000 years ago, during the late Pleistocene Epoch. Anastomosing flood channels, giant current ripples, bergmounds, and giant flood bars are among the landforms created by the floods and are readily seen on the Hanford Site. Most of the...
large landslides in the region occurred when these flood waters eroded the steep slopes of the ridges. The 200 Areas are located on a major Pleistocene flood bar, the Cold Creek bar.

Since the end of the Pleistocene, winds have locally reworked the flood sediments, depositing dune sands in the lower elevations and loess (windblown silt) around the margins of the Pasco Basin. Generally sand dunes have been stabilized by anchoring vegetation except where they have been reactivated where vegetation is disturbed.

1.4.3.2 Stratigraphy

The stratigraphy of the Pacific Northwest is a collection of accreted terrane rocks added onto the North American craton between the Precambrian and early Cenozoic. The accreted terrane rocks are now mostly covered by extensive Cenozoic volcanic rocks. Intercalated with the volcanic rocks in structural basins and along the evolving continental margin are sedimentary rocks derived from eroding uplands areas.

The generalized stratigraphy of the Columbia Basin and Hanford Site is shown in Figure 1-20. The principal bedrock of the Hanford Site is the CRBG and intercalated and overlying sedimentary rocks of the Ellensburg Formation. The overlying sedimentary rocks are principally of the Ringold Formation and the Pleistocene catastrophic flood deposits of the Hanford formation (informal designation). This is the general stratigraphy present at the SNF Project facilities.

Rocks older than the CRBG are exposed primarily along the margins of the Columbia Basin and vary widely in age, lithology, and structure. It is this complex stratigraphy and structure that underlies the CRBG of the Columbia Basin and Hanford Site and that is the probable source for seismic hazards under the basalt. Because of the thick basalt cover, the exact nature of the stratigraphy and the types of the structure under the Hanford Site is uncertain.

Along the northern and northwestern margins of the Columbia Basin in early Tertiary time, a series of sedimentary basins formed in the accreted terranes of the North Cascades (DOE/RW-0164, Campbell 1989). These basins are now separated by tectonic "blocks" or uplifts that have a northeast- to northwest-trending structural grain and trend toward the Hanford Site. Along the northeastern and eastern margins of the Columbia Basin, the CRBG laps onto Paleozoic rocks and Precambrian miogeosynclinal and cratonic rocks that are interspersed with granites of the Idaho Batholith and other Jurassic and Cretaceous intrusions (DOE/RW-0164). To the south and southwest, lower- to middle-Tertiary volcanic rocks and related volcanioclastic rocks primarily of the Clarno and John Day Formations directly underlie the CRBG and overlie accreted terranes of the Blue Mountains. To the west, younger volcanic rocks that erupted from the High Cascades cover the CRBG and obscure early Cenozoic volcanic and volcanioclastic rocks of predominantly Eocene and Oligocene age.

1.4.3.2.1 Columbia River Basalt Group

The CRBG, the principal bedrock unit in the Yakima Fold Belt (YFB) and under the Hanford Site, is a sequence of tholeitic flood basalt flows that were erupted between 17 and 6 million years ago (Ma) (Figure 1-20). The CRBG now covers approximately 63,325 mi² and consists of 41,700 mi³ of basalt (Tolan et al. 1989). The
flows were erupted from north-northwest-trending fissures or linear vent systems in north-central and northeastern Oregon, eastern Washington, and western Idaho (Swanson et al. 1975).

The CRBG has been divided into five formations (Swanson et al. 1979), only the Grande Ronde Basalt, the Wanapum Basalt, and the Saddle Mountains Basalt are exposed on the Hanford Site. The Imnaha Basalt occurs at the base of the CRBG under the Hanford Site (Tolan et al. 1989).

The basalt flows of the CRBG are recognized using a combination of lithology, chemistry, and paleomagnetic data (Swanson et al. 1979). Chemical composition and paleomagnetic data have proven to be the most reliable criteria for flow recognition and correlation. Lithology is reliable for many flows primarily within the Wanapum and Saddle Mountains Basalts, but chemical compositions are still used to confirm identifications.

Over 85% of the CRBG was erupted during the 2-million-year span of the Grande Ronde Basalt eruptions. In the field, the Grande Ronde Basalt is divided into four magnetostratigraphic units, which, from oldest to youngest are Reversed 1 (R1), Normal 1 (N1), Reversed 2 (R2), and Normal 2 (N2) (Swanson et al. 1979). The Grande Ronde Basalt is further subdivided into seventeen groups of flows based on chemical compositions (Reidel et al. 1989a).

The younger basalt flows of the Wanapum and Saddle Mountains Basalts on the Hanford Site have been locally eroded to varying degrees. Uplift along anticlinal ridges has resulted in erosion to different depths along the margins of the Pasco Basin. Within the synclines where the basalt surface is covered by sediment fill, the upper basalt flows have been locally eroded by fluvial activity and proglacial flooding. North of the 200 Areas between Gable Mountain and Gable Butte, the Saddle Mountains Basalt has been eroded down to the oldest member, the Umatilla Member. The Elephant Mountain Member is suspected of being eroded near the northeast corner of 200 East Area.

14322 Intercalated and Post-Columbia River Basalt Group Sediments The Hanford Site and the SNF Project facilities are situated on a sequence of sedimentary units that overlie the CRBG. These sediments are confined largely to the synclinal valleys and basins of the western Columbia Basin. The sedimentary record is incomplete, but it is a direct reflection of the structural development of the area (Fecht et al. 1987). The upper-Miocene to middle-Pliocene (late Neogene) record of the Columbia River system in the Columbia Basin is represented by the upper Ellensburg Formation, Ringold Formation, and Snipes Mountain conglomerate. Except for local deposits (e.g., the Pliocene-Pleistocene unit and the early Palouse soil [DOE/RW-0164]), there is a hiatus (i.e., lack of sedimentation or erosion) in the stratigraphic record between the end of the Ringold (3.4 Ma) and the Pleistocene (1.6 Ma).

Pleistocene to Recent (Quaternary) sediments overlying the CVDF at the Hanford Site include cataclysmic flood gravels and slackwater sediments of the Hanford formation.
Late Neogene Sediments Two main late Neogene sedimentary units occur in the
Columbia Basin and at the Hanford Site, the Ellensburg Formation and the Ringold Formation
(Figure 1-20)

Ellensburg Formation The Ellensburg Formation includes epiclastic and volcanoclastic
sedimentary rocks that are intercalated with and overlie the CRBG (Waters 1961, 
Swanson et al. 1979) The Ellensburg stratigraphy of the Hanford Site has been discussed in
RHO-BWI-ST-14, Subsurface Geology of the Cold Creek Syncline, in Geological Society of 
America Special Paper 239, Volcanism and Tectonism in the Columbia River Flood-Basalt
Province (Reidel et al. 1989a), and in Bulletin 77, Selected Papers on the Geology of Washington
(Fecht et al. 1987) At the Hanford Site, the Ellensburg Formation is mixed with sediments
deposited by the ancestral Clearwater River and Columbia River

Ringold Formation Sediments continued to be deposited in most synclinal valleys of the
central Columbia Basin including the Hanford Site, long after the eruptions of the CRBG these
sediments are the Ringold Formation Although exposures of the Ringold Formation are limited
to the White Bluffs on the east side of the Hanford Site, isolated exposures on the west side of the
Hanford Site, and the Smyrna and Taunton Benches within the Othello Basin, extensive data on
the Ringold Formation have been obtained from boreholes at the Hanford Site

The Ringold Formation at the Hanford Site is up to 607 ft thick in the deepest part of the
Cold Creek syncline south of the 200 West Area and 558 ft thick in the western Wahluke syncline
near the 100 B Area The Ringold Formation pinches out against the Gable Mountain,
Yakima Ridge Saddle Mountains, and Rattlesnake Mountain anticlines This formation is present
under the CSB site

The Ringold Formation consists of semi-indurated clay, silt, pedogenically altered sediment,
fine- to coarse-grained sand, and granule to cobble gravel Ringold strata typically are below the
water table and the textural variations of the strata influence groundwater flow The Ringold
Formation historically has been divided into a variety of units, facies types and cycles
(Newcomb 1958, RHO-BWI-ST-4 RHO-ST-23 SD-BWI-DP-039 and DOE/RW-0164)
However these terminologies have proven to be of limited use because they are too generalized
to account for significant local stratigraphic variation or because they were defined in detail for
relatively small areas and do not account for basinwide stratigraphic variation
(WHC-SA-0740-FP WHC-SD-EN-EE-004)

Recent studies of the Ringold Formation in the Hanford Area indicate it contains significant
stratigraphic variations (WHC-SA-0740-FP WHC-SD-EN-EE-004) that are best described on
the basis of sediment facies Sediment facies in the Ringold Formation are defined on the basis of
lithology, stratification, and pedogenic alteration

- The fluvial gravel facies consists of clast-supported granule to cobble gravels with a
sandy matrix and intercalated sands and muds Clast composition is variable but
typically includes basalt quartzite, porphyritic volcanics and greenstone Sands
generally are quartzo-feldspathic with less than 25% basalt content Bedforms have
low angle to planar stratification, massive bedding, wide shallow channels, and large-scale cross-bedding. The facies was deposited in a gravelly fluvial braidplain characterized by wide, shallow, shifting channels.

- The fluvial sand facies consists of quartz-feldspathic sands, cross-bedded and cross-laminated sands that are intercalated with lenticular silty sands, clays, and thin gravels. These sands usually contain less than 15% basaltic lithic fragments, and fining upwards sequences are common. Strata of this type were deposited in wide, shallow channels.

- The overbank facies consists of laminated to massive silt, silty fine-grained sand, and paleosols containing variable amounts of pedogenic calcium carbonate. Overbank deposits occur as thin lenticular interbeds in the gravels and sands and as thick laterally continuous sequences. These sediments record deposition in proximal levee to more distal floodplain conditions.

- The lacustrine facies is characterized by plane laminated to massive clay with thin silt and silty sand interbeds displaying some soft-sediment deformation. Deposits coarsen upwards. Strata were deposited in a lake under standing water to deltaic conditions.

- The alluvial fan facies is characterized by massive to crudely stratified, weathered to unweathered basaltic detritus. These deposits generally are found around the periphery of the Pasco Basin and record deposition by debris flows in alluvial fan settings and in sidestreams draining into the Pasco Basin.

The lower half of the Ringold Formation is the main unconfined aquifer under the Hanford Site and contains five separate stratigraphic intervals dominated by the fluvial gravels facies. These gravels, designated units A, B, C, D, and E (Figure 1-21), are separated by intervals containing deposits typical of the overbank and lacustrine facies (WHC-SD-EN-EE-004). The lowermost of the fine-grained sequences overlying unit A is designated the Lower Mud sequence. The uppermost gravel unit, unit E, grades upwards into interbedded fluvial sand and overbank deposits that are in turn overlain by lacustrine-dominated strata. Fluvial gravel units A and E correspond to the lower basal and middle Ringold units respectively as defined in DOE/RW-0164, Site Characterization Plan Reference Repository Location Hanford Site Washington. Gravel units B, C, and D do not correlate to any previously defined units (WHC-SD-EN-EE-004). The Lower Mud sequence corresponds to the upper basal unit and lower unit as defined by DOE/RW-0164.

The upper part of the Ringold Formation consists of the sequence of fluvial sands, overbank deposits, and lacustrine sediments overlying unit E. These sediments correspond to the upper unit as originally defined in "Ringold Formation of the Pleistocene Age in the Type Locality, The White Bluffs, Washington," (Newcomb 1958) along the White Bluffs in the eastern Pasco Basin. The fluvial sand facies is the principal facies of the upper part under the CSB.
The CRBG lies approximately 527 ft beneath the 100 K Area. Overlying the CRBG is the Ringold Formation.

The Ringold Formation is about 493 ft thick and consists of the fluvial gravels and sands of units A, B, C, and E, lacustrine and fluvial overbank deposits, and palesols (WHC-SD-EN-TI-011) (Figure 1-21). Unit A, the lowermost unit, is approximately 23 ft thick and consists of fluvial gravel facies grading upward into sand associations. Overlying unit A is the Lower Mud unit. The Lower Mud unit is approximately 105 ft thick. Unit B is approximately 92 ft thick and consists predominantly of sand. Overlying unit B is a 209-ft-thick sequence of muds and sandy muds, typically displaying characteristics of palesols and fluvial overbank deposits. The sequence has three parts: an upper and a lower part that are predominantly silt to sandy silt, and a middle section of gravelly sand. The uppermost unit of the Ringold Formation at the 100 K Area is the coarse-grained unit E, which is predominantly composed of the fluvial gravel and fluvial sand facies. The unconfined portion of the uppermost aquifer system occurs in this unit at the 100 K Area. The known thickness of unit E ranges from 64 to 141 ft.

The Hanford formation at the 100 K Area is a wedge that decreases in thickness toward the Columbia River. It is approximately 120 to 130 ft thick. The Hanford formation pinches out from southeast to northwest across the 100 K Area. The gravel-dominated facies predominates in the Hanford formation throughout the 100 K Area. Boulder gravel is often found in the upper 20 to 50 ft. The sand-dominated facies occurs locally in a few intervals, but it is not thick enough or extensive enough to correlate from borehole to borehole (Figure 1-22). The silt-dominated facies has not been identified in the 100 K Area.

Holocene deposits in the study area are dominated by Columbia River deposits and eolian deposits. Columbia River deposits consist of gravels and coarse-grained sands deposited in channels and silts and fine sands deposited in overbank area. Eolian deposits consist dominantly of less than 3 ft of silty, fine-grained sands that blanket much of the area except in locations where they were removed for construction purposes. In many locations, eolian deposits are only a thin blanket (<1 ft).

The 100 K Area is geologically different than surrounding areas (100 B, 100 C, and 100 N Areas) because the Ringold Formation is exposed, not only along the banks of the Columbia River, but also from the river to 1200 ft or more away from the river to the southeast.

Quaternary Stratigraphy of the Pasco Basin. Quaternary sediments as much as 328 ft thick within the Pasco Basin overlie the Ringold Formation. The most extensive of these is the Pleistocene-aged Hanford formation, which is about 50 ft thick at the K Basins and CVDF (Figure 1-21). Locally the Hanford formation and underlying Ringold Formation are separated by several laterally discontinuous and informally defined units. These units are the Plio-Pleistocene unit, the pre-Missoula gravels, and the early Palouse soil (Figure 1-21) but they are absent at the SNF Project facilities.

Quaternary Alluvium. Fluvial deposits from major rivers (Yakima, Snake, Columbia) are represented in the Pasco Basin ranging in age from Tertiary to present (Baker et al 1991). There
are two main alluvial units that are recognized at the Hanford Site: the Plio-Pleistocene unit and the pre-Missoula gravels. Neither of these units is present at the K Basins and CVDF. See Section A1 4.3 in Annex A for discussions related to the CSB.

Overlying the tilted and truncated Ringold Formation in an unconformable relationship in the western Cold Creek syncline in the vicinity of 200 West Area is the laterally discontinuous Plio-Pleistocene unit (DOE/RW-0164). The Plio-Pleistocene unit appears to be correlative to other sidestream alluvial and pedogenic deposits found near the base of the ridges bounding the Pasco Basin on the north, west, and south. These sidestream alluvial and pedogenic deposits are inferred to have a late Pliocene to early Pleistocene age on the basis of stratigraphic position and the magnetic polarity of interfingering loess units.

In the central Pasco Basin, mainstream alluvium lies stratigraphically between the Ringold Formation and the Hanford formation. A thick sheet of well-rounded and well-sorted quartzose to gneissic clast-supported pebble to cobble gravel with a quartzo-feldspathic sand matrix gravel, informally called the "pre-Missoula gravels," overlies the Ringold Formation in much of the central Pasco Basin. Based on magnetic polarity and stratigraphic position, this unit is interpreted to be early Pleistocene. Mainstream alluvium of probable early Pleistocene age is exposed along Cold Creek and the Yakima Bluffs. It is unclear whether the pre-Missoula gravels overlie or interfinger with the early Palouse soil and Plio-Pleistocene unit. The pre-Missoula gravels are interpreted as mainstream deposits in the Columbia River.

**Eolian Deposits.** Loess deposits at Hanford contain a detailed Quaternary record. Five units are represented within the Pasco Basin (WHC-MR-0391). These units are informally referred to as L1 through L5 and differentiated on the basis of position relative to other stratigraphic units, color, soil development, and paleomagnetic polarity.

The main eolian unit in the subsurface at the Hanford Site is referred to as the early Palouse soil. The early Palouse soil consists of up to 66 ft of massive, brown-yellow, compact loess-like silt and minor fine-grained sand (RHO-ST-23, DOE/RW-0164). Granule-sized grains consisting primarily of basalt are common in this unit. These deposits overlie the Plio-Pleistocene unit (Figure 1-21) in the western Cold Creek syncline around the 200 West Area but are absent in the 200 East Area. The unit is differentiated from overlying graded rhythmites (Hanford formation) by greater calcium carbonate content, massive structure, and high natural gamma response in geophysical logs (DOE/RW-0164). The upper contact of the unit is poorly defined and it may grade up-section into silty strata commonly found in the lower part of the Hanford formation. Because of a predominantly reversed polarity, the unit is inferred to be early Pleistocene in age.

**Hanford Formation.** The Hanford formation is the informal name given to all cataclysmic flood deposits of the Pleistocene. This is the main stratigraphic unit at the surface of the SNF Project facilities. The Hanford formation is thickest in the vicinity of 200 West and 200 East Areas where it is up to 200 ft thick. Hanford deposits are absent on ridges above approximately 1250 ft above sea level, the highest level of cataclysmic flooding in the Pasco Basin (RHO-BW-SA-563A).
The Hanford formation consists of pebble to boulder gravel, fine- to coarse-grained sand and silt. These deposits are divided into three facies: gravel-dominated, sand-dominated, and silty. These facies are referred to as coarse-grained deposits, plane-laminated sand facies, and rhythmite facies, respectively, in RHO-BW-SA-563A, *Quaternary Geology of the Pasco Basin, Washington*. The rhythmite facies are also referred to as the "Touchet Beds." The Hanford formation is thickest in the vicinity of 200 West and 200 East Areas, where it is up to 200 ft thick. Hanford deposits are absent on ridges above approximately 1,250 ft above sea level, the highest level of catastrophic flooding in the Pasco Basin (RHO-BW-SA-563A).

- The gravel-dominated facies generally consists of coarse-grained basaltic sand and granule to boulder gravel. These deposits display massive bedding, plane to low-angle bedding, and large-scale planar cross-bedding in outcrop. The gravel facies dominates the Hanford formation in the 100 Areas north of Gable Mountain. The gravel-dominated facies was deposited by high-energy flood waters in or immediately adjacent to the main catastrophic flood channelways.

- The sand-dominated facies consists of fine- to coarse-grained sand and granule gravel displaying plane lamination and bedding and less commonly plane bedding and channel-fill sequences in outcrop. This facies is most common in the central Cold Creek syncline and in the central to southern parts of the 200 East and 200 West Areas. The laminated sand facies was deposited adjacent to main flood channelways during the waning stages of flooding. The facies is transitional between the gravel-dominated facies and the rhythmite facies.

- The silty facies consists of thinly bedded, plane laminated, and ripple cross-laminated silt and fine- to coarse-grained sand that commonly displays normally graded rhythms a few centimeters to several tens of centimeters thick (RHO-BWI-ST-4 DOE/RW-0164). This facies is found throughout the central, southern, and western Cold Creek syncline within and south of the 200 East and West Areas. These sediments were deposited under slackwater conditions and in back-flooded areas (DOE/RW-0164).

**Volcanic Ash Deposits** Volcanism in the Cascade Range has been active throughout the Pleistocene Epoch (approximately 2 million years before present to 10,000 years before present) and the Holocene Epoch (10,000 years before present to present). The eruption history of the Holocene best characterizes the most likely types of activity in the next 100 years. Many of the volcanoes in the Cascade Range have been active in the last 10,000 years, including Mount Mazama (Crater Lake) and Mount Hood in Oregon and Mount St. Helens, Mount Adams, and Mount Rainier in Washington State. Quaternary sediments recorded these eruptions in the form of ash deposits that are interlayered with the sediments.

The Hanford Site is approximately 95 mi from Mount Adams, 110 mi from Mount Rainier, and 125 mi from Mount St. Helens, the three closest active volcanoes. At these distances, the tephra (ash) is the only volcanic product to reach the Site and, thus, the only volcanic hazard at the Hanford Site (Figure 1-23). Mount St. Helens has been considerably more active throughout...
the Holocene than Mount Rainier or Mount Adams, which is the least active of the three. Less
than 0.4 in of ash was deposited on parts of the Hanford Site during the 1980 Mount St Helens
eruption.

1433 History of Cataclysmic Flooding in the Pasco Basin Cataclysmic floods inundated the
Pasco Basin several times during the Pleistocene when ice dams failed in northern Washington.
Net erosion by these floods was minimal and probably associated with only the earliest floods,
later floods only partially incised into older flood deposits before backfilling. During the three
major flood episodes there were probably numerous individual flood events. Deciphering the
history of cataclysmic flooding in the Pasco Basin is complicated, not only by floods from multiple
sources but also because the paths of Missoula floodwaters migrated and changed course with
the advance and retreat of the Cordilleran Ice Sheet. The best preserved record is that of the last
Missoula flood, which apparently came down the Columbia River. The uppermost 50 ft of
sediments at the 100 K Area are flood deposits.

1434 Geologic Structures of the Columbia Basin and Hanford Site The geologic structure
of the Pacific Northwest is controlled by a basement rock assemblage of accreted terranes fused
onto the structurally complex North American craton from the early Mesozoic to early Cenozoic.
The accreted terranes form the backbone of the Cascade Range, Okanogan Highlands, and the
Blue Mountains. The terranes are now mostly covered by extensive cover of Cenozoic rocks that
were folded and faulted in a north-south oriented compressive regime. North-south compression
is continuing today east of the Cascades portion of the Pacific Northwest and this pattern of
Cenozoic deformation is expected to continue into the future.

The Columbia Basin is a structurally and topographically low area surrounded by mountains
ranging in age from the late Mesozoic to Recent (May 5-18). The Columbia Basin has two
major structural subdivisions or subprovinces: the YFB and the Palouse Slope (Figure 1-18).
The YFB is a series of anticlinal ridges and synclinal valleys in the western and central parts of
the Columbia Basin. The Palouse Slope forms the eastern part of the Columbia Basin and is mainly
a westward-tilting paleoslope. The Hanford Site and the CVDF are in the eastern part of the YFB.
The western boundary of the Palouse Slope is at the eastern boundary of the Hanford Site.

The Blue Mountains subprovince of the Columbia River flood-basalt province is a
northeast-trending anticlinorium that extends 155 mi from the Oregon Cascades to Idaho and
forms the southern border of the Columbia Basin and the southern part of the Columbia Plateau.

14341 Major Structural Features of the Columbia Basin The main structural
features of the Columbia Basin are the YFB and two major cross-cutting structures: the Olympic
Wallowa lineament (OWL) and the Hog Ranch-Naneum Ridge anticline (HR-NR). The OWL
passes along the southern boundary of Hanford and the HR-NR anticline forms the western
structural boundary of the Pasco Basin and Hanford Site.

The Olympic-Wallowa Lineament The OWL (Figure 1-24) is a major topographic
feature in Washington and Oregon that crosscuts the Columbia Basin and forms the southern
boundary of the Hanford Site. This alignment of structural features parallels prebasalt structural
trends along the northwest margin of the Columbia Basin, but it has not been linked to any individual structure (Campbell 1989; Reidel and Campbell 1989).

Northwest of the CRBG margin, numerous northwest- and north-trending faults and shear zones of the Strait Creek fault system lie subparallel to the OWL. The Snoqualmie batholith intrudes these faults but is not cut by them, indicating that any possible movement along the OWL at the western margin of the Columbia Basin must be older than the batholith, 17 to 19.7 Ma (DOE/RW-0164).

The structural significance of the OWL has been called into question by two recent geophysical studies. Neither a seismic profiling survey (Jarchow 1991) nor a gravity survey (Saltus 1993) could find any obvious geophysical signature for the OWL below the CRBG.

**Hog Ranch-Naneum Ridge Anticline** The NR-HR anticline is the western structural boundary of the Pasco Basin, the basin containing the Hanford Site (Figure 1-24). The HR-NR anticline is a broad south-trending anticline in the CRBG that crosses the YFB in a north-south direction. This south-plunging structure passes through five Yakima folds and the OWL. The HR-NR anticline was active in late to middle Miocene as demonstrated by thinning of basalt flows across it (Reidel et al. 1989b), but the east-trending Yakima folds show no apparent offset by the cross structure (Campbell 1989; Keinle et al. 1977; Reidel et al. 1989b), nor is the HR-NR anticline offset where the OWL-Cle Elum-Wallula deformed zone (CLEW) crosses it. Growth of the HR-NR anticline continued from the Miocene to Recent and is now marked by the highest structural points along the ridges that cross it.

**The Yakima Fold Belt** The YFB subprovince covers about 5,400 mi² of the western Columbia Basin (Figures 1-19 and 1-24) and formed as basalt flows and intercalated sediments were folded and faulted under north-south directed compression. The reader is referred to Tables 1-17 and 1-18 and to the compilation of structural features for the Columbia Basin by Tolan and Reidel (1989). The Hanford Site lies in the Pasco Basin, which is one of the larger structural basins near the eastern limit of the YFB. The YFB gives the Hanford Site its principal physiographic features and has been the primary factor influencing the geohydrologic conditions at the Hanford Site.

The YFB consists of asymmetrical anticlinal ridges and synclinal valleys. The anticlines are typically segmented and usually have a north vergence, although some folds have a south vergence. Synclines are typically asymmetrical with a gently dipping north limb and a steeply dipping south limb. Fold length is variable ranging from several miles to over 62 mi; fold wavelengths range from several miles to as much as 12 mi. Structural relief is typically about 2,000 ft but varies along the length of the fold. The greatest structural relief along the Frenchman Hills, the Saddle Mountains, Umtanum Ridge, and Yakima Ridge occurs where these folds intersect the north-south trending HR-NR anticline (Reidel et al. 1989b).
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<th>Length/width (km)</th>
<th>Amplitude (maximum)</th>
<th>Trend</th>
<th>Number of segments</th>
<th>Segment length (x=mean σ=1 standard deviation)</th>
<th>Vergence</th>
<th>Amount of sh rtemng (best approx.)</th>
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<td>90 to 100°</td>
<td>7</td>
<td>x=14 σ=10 R=7 to 35</td>
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<tr>
<td>Umtanum Ridge</td>
<td>110 km/3 10 km</td>
<td>520 m</td>
<td>90 to 130°</td>
<td>9</td>
<td>x=11 σ=4.2 R=5 to 17</td>
<td>N</td>
<td>1.3 km</td>
<td>Asymmetrical tight to open echelon segments on east end</td>
</tr>
<tr>
<td>Cleman Mountain</td>
<td>35 km/8 km</td>
<td>950 m</td>
<td>130</td>
<td>2</td>
<td>x=18 σ=8 R=13 to 23</td>
<td>S</td>
<td>NK &gt;1 km</td>
<td>Asymmetrical</td>
</tr>
<tr>
<td>Yakima Ridge</td>
<td>100 km/5 10 km</td>
<td>550 m</td>
<td>135 to 225</td>
<td>12</td>
<td>x=12 σ=8 R=5 to 30</td>
<td>N</td>
<td>NK &gt;3 km</td>
<td>Asymmetrical gentle to open echelon segments box fold segments</td>
</tr>
<tr>
<td>Rattlesnake Mtn and Rattles</td>
<td>85 km/5 20 km</td>
<td>800 m</td>
<td>310</td>
<td>11</td>
<td>x=9 σ=6 R=5 25</td>
<td>N</td>
<td>NK &gt;3 km</td>
<td>Asymmetrical tight to open faulted out hinge doubly plunging</td>
</tr>
<tr>
<td>Rattlesnake Ahtanum Ridge</td>
<td>100 km/5 8 km</td>
<td>610 m</td>
<td>238 to 108</td>
<td>11</td>
<td>x=9 σ=4 R=5 to 18</td>
<td>N</td>
<td>NK &gt;1 km</td>
<td>Asymmetrical gentle to open</td>
</tr>
<tr>
<td>Toppenish Ridge</td>
<td>85 km/4 8 km</td>
<td>500 m</td>
<td>118 to 258°</td>
<td>5</td>
<td>x=17 σ=7 R=10 to 28</td>
<td>N</td>
<td>NK &gt;1 km</td>
<td>Asymmetrical tight to open</td>
</tr>
<tr>
<td>Snipes Mountain</td>
<td>13 km/1 km</td>
<td>150 m</td>
<td>110</td>
<td>3</td>
<td>13 km</td>
<td>S</td>
<td>NK &lt;1 km</td>
<td>Asymmetrical tight to open</td>
</tr>
<tr>
<td>Anticline</td>
<td>Length/width (km)</td>
<td>Amplitude (maximum)</td>
<td>Trend</td>
<td>Number of segments</td>
<td>Segment length ( \bar{x}=17 ) ( \sigma=5 ) ( R=5 ) to 20</td>
<td>Vergence</td>
<td>Amount of shortening (best approx)</td>
<td>Geometry</td>
</tr>
<tr>
<td>----------------------</td>
<td>-------------------</td>
<td>---------------------</td>
<td>-------</td>
<td>-------------------</td>
<td>-------------------------------------------------</td>
<td>----------</td>
<td>---------------------------------</td>
<td>-----------------------------------------------</td>
</tr>
<tr>
<td>Horse Heaven Hills</td>
<td>185 km/5 30 km E 2 7 km W</td>
<td>335 1100 m</td>
<td>115 to 255</td>
<td>21</td>
<td>( x=17 ) ( \sigma=5 ) ( R=5 ) to 20</td>
<td>N</td>
<td>&gt; 2 km (0.67 to 1.25 km, 117% from folding)</td>
<td>Asymmetrical tight to open, en echelon subsidiary crest folds box folds</td>
</tr>
<tr>
<td>Columbua Hills</td>
<td>170 km/5 10 km</td>
<td>250 365 m</td>
<td>255</td>
<td>10</td>
<td>( x=15 ) ( \sigma=6 ) ( R=6 ) 23</td>
<td>S</td>
<td>NK &gt; 2 km</td>
<td>Asymmetrical tight to open doubly plunging en echelon subsidiary crest folds box folds</td>
</tr>
</tbody>
</table>

NK = not known
Table 1-18 Characteristics of Major Faults

<table>
<thead>
<tr>
<th>Fault zone</th>
<th>Length</th>
<th>Trend</th>
<th>Horizontal offset</th>
<th>Vertical offset (best approx)</th>
<th>Dip and fault direction</th>
<th>Age of last movement</th>
</tr>
</thead>
<tbody>
<tr>
<td>CLEW</td>
<td>290 km</td>
<td>310</td>
<td>0-4 km</td>
<td>0-800 m</td>
<td>Reverse</td>
<td>Quaternary</td>
</tr>
<tr>
<td>RAW (includes Wallula Fault zone)</td>
<td>125 km</td>
<td>310</td>
<td>0-4 km</td>
<td>0-800 m</td>
<td>Reverse</td>
<td>Quaternary</td>
</tr>
<tr>
<td>Hite Fault system</td>
<td>135 km</td>
<td>330</td>
<td>NK</td>
<td>NK 0-900 m</td>
<td>Vertical en echelon and strike slip</td>
<td>Recent 1936 Milton Freewater earthquake</td>
</tr>
<tr>
<td>Frenchman Hills</td>
<td>100+ km</td>
<td>270</td>
<td>&gt;300 m</td>
<td>~200 m</td>
<td>&gt;45 S</td>
<td>&gt;500 000 years</td>
</tr>
<tr>
<td>Saddle Mountains</td>
<td>100 km</td>
<td>270</td>
<td>&gt;2.5 km</td>
<td>600 m</td>
<td>&gt;60 S</td>
<td>&gt;3.4 Ma</td>
</tr>
<tr>
<td>Manastash Hansen Creek</td>
<td>70 km</td>
<td>300</td>
<td>&lt;1 km</td>
<td>~300 m</td>
<td>Reverse thrust</td>
<td>&gt;1.34 Ma</td>
</tr>
<tr>
<td>Umtanum</td>
<td>110 km</td>
<td>270</td>
<td>310</td>
<td>&gt;300 m</td>
<td>1500 m</td>
<td>30 70 S</td>
</tr>
<tr>
<td>Cleman Mountain</td>
<td>20? km</td>
<td>310</td>
<td>NK</td>
<td>~900 m</td>
<td>Reverse thrust N</td>
<td>NK</td>
</tr>
<tr>
<td>Yakima Ridge</td>
<td>120+ km</td>
<td>225</td>
<td>to S45 E</td>
<td>NK  ~500 m</td>
<td>Reverse thrust S</td>
<td>~&lt;1 Ma</td>
</tr>
<tr>
<td>locally N</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Rattlesnake Ahtanum Ridge</td>
<td>100 km</td>
<td>558</td>
<td>W to 315</td>
<td>~800 m</td>
<td>Reverse thrust S</td>
<td>~13 000 years</td>
</tr>
<tr>
<td>Toppensh Ridge</td>
<td>65 90 km</td>
<td>258</td>
<td>to 298 W</td>
<td>~500 m</td>
<td>Reverse thrust S</td>
<td>Recent</td>
</tr>
<tr>
<td>Horse Heaven Hills</td>
<td>200+ km</td>
<td>245</td>
<td>to 295 W</td>
<td>~335 1100 m</td>
<td>Reverse thrust S</td>
<td>NK</td>
</tr>
<tr>
<td>Columbia Hills</td>
<td>160 km</td>
<td>245</td>
<td>NK &gt;1 km</td>
<td>~365 m</td>
<td>70 N</td>
<td>NK</td>
</tr>
<tr>
<td>Northwest trending faults</td>
<td>40 120 km</td>
<td>320</td>
<td>&lt;100 m</td>
<td>&lt;100 m</td>
<td>Strike slip vertical (dip reversal)</td>
<td>Holocene</td>
</tr>
</tbody>
</table>

CLEW = Cle Elum Wallula (deformed zone)
Ma = million years ago
NK = not known
R = radius
RAW = Rattlesnake Mountain to Wallula Gap

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In general, the axial trends produce a "fanning" pattern across the fold belt (Figure 1-24). Anticlines on the western side of the fold belt generally have a N 50° E trend (Open File Report 79-1363). Anticlines in the central part of the fold belt have east-west trends except along the OWL where a N 50° W trend predominates.

Within the Hanford Site and surrounding area, the geometry of the anticlines typically consists of steeply dipping to overturned north flanks and gently dipping (< 5°) south flanks. Exceptions, however, include the doubly plunging anticlines within the Rattlesnake-Wallula alignment of the CLEW and the conjugate box-fold geometry of parts of the anticlines such as the Smyrna segment of the Saddle Mountains (Reidel 1984). The main variable in fold profiles is the width of the gently dipping limb. The widths of the gently dipping limbs vary from as little as 3 mi to as much as 22 mi.

Segmentation of the anticlines is common throughout the fold belt and is defined by abrupt changes in fold geometry or by places where regional folds die out and become a series of doubly plunging anticlines. Segment lengths are variable but average about 7 mi (ranging from 3 to 22 mi) near the Hanford Site, some of the larger segments contain subtler changes in geometry, such as different amplitudes, that could also be considered segment boundaries. Segment boundaries are often marked by cross or tear faults that trend N 20° W to north and display a principal component of strike-slip movement (e.g., Saddle Mountains [Reidel 1984]). Near Hanford these cross faults are confined to the anticlinal folds and usually occur only on the steeper limb, dying out onto the gentler limb.

Segment boundaries may also be marked by relatively undeformed areas along the fold trend where two fold segments plunge toward each other. For example, the Yakima River follows a segment boundary where it crosses the Rattlesnake-Wallula alignment at the southeast termination of Rattlesnake Mountain (Figure 1-24).

The steep limb of the asymmetrical anticlines is almost always faulted. Near the Hanford Site the steep limb is typically the northern flank, but elsewhere as at the Columbia Hills (Open File Report 79-1363) the south limb is faulted. Where exposed, these frontal fault zones have been found to be imbricated thrusts as, for example, at Rattlesnake Mountain, Umtanum Ridge near Priest Rapids Dam (Bentley in Open File Report 79-1363) the Horse Heaven Hills (RHO-BWI-SA-344 P) and the Saddle Mountains near Sentinel Gap (Reidel 1984).

Yakima folds have emergent thrust faults at the ground surface. The tops of the youngest lava flows at the earth's surface serve as a plane that becomes a low angle thrust fault, the structural attitude of the surface flow controls the angle of the emergent fault plane. This type of apparent structural control led many investigators to conclude that faults associated with the Yakima Folds are low-angle thrust faults with detachment surfaces either within the CRBG, in the sediments below the basalts or at the basalt-sediment contact. Where erosion provides deeper exposures into the cores of folds, the frontal faults are observed to be reverse faults (e.g., the Columbia water gap in the Frenchman hills, 45° south [Grolier and Bingham 1978], the Columbia Hills at Rock Creek, Washington, 50°-70° north [Open File Report 79-1363]).
Hydrocarbon exploration boreholes provide direct evidence for the dips of these frontal faults. Reidel et al. (1989b) have shown that the Saddle Mountains fault must dip more than 60° where the Shell-ARCO BN 1-9 borehole was drilled. Drilling of the Umtanum fault near Priest Rapids Dam (DOE/RW-0164) suggests that this fault dips southward under the ridge with a dip of at least 30° to 40° but perhaps as high as 60° (Price and Watkinson 1989).

Although it is difficult to assess, total shortening increases from east to west across the YFB. At about 120° longitude, it is estimated to be greater than 9 mi but less than 16 mi (Reidel et al. 1989b) or about 5% (Table 1-17). Typically, shortening on an individual anticline caused by folding is approximately 0.5 to 1.0 mi. The amount of shortening on faults expressed at the surface is generally unknown. Estimates range from several hundreds of meters to as much as 1.9 mi (Table 1-18).

14342 Structure of the Hanford Site. The Cold Creek syncline (Figure 1-25) lies between the Umtanum Ridge-Gable Mountain uplift and the Yakima Ridge uplift, and is an asymmetric and relatively flat-bottomed structure. The 200 Areas lie on the northern flank, and the bedrock dips gently (approximately 5°) to the south. The 300 Area lies at the eastern end of the Cold Creek syncline where it merges with the Pasco syncline.

The Wahluke syncline (Figure 1-25) is the principal structural unit that contains the 100 Areas and the CVDF. The Wahluke syncline is an asymmetric and relatively flat-bottomed structure similar to the Cold Creek syncline. The northern limb dips gently (approximately 5°) to the south. The steepest limb is adjacent to the Umtanum Ridge-Gable Mountain structure.

The Umtanum Ridge-Gable Mountain structural trend is a segmented anticlinal ridge extending for a length of 68 mi in an east-west direction. It passes north of the 200 and 300 Areas and south of the 100 Areas. From the west, the Umtanum Ridge plunges eastward and joins the Gable Mountain-Gable Butte segment just east of the western boundary of the Hanford Site. The easternmost segment, the Southeast anticline, trends southeast off the eastern boundary of the Gable Mountain-Gable Butte segment. Gable Mountain and Gable Butte are two topographically isolated anticlinal ridges that are composed of a series of northwest trending, doubly plunging, en echelon anticlines, synclines and associated faults.

The Yakima Ridge uplift extends from west of Yakima, Washington, to the center of the Pasco Basin, where it forms the southern boundary of the Cold Creek syncline south of 200 West Area (Figure 1-25). The easternmost surface expression of the Yakima Ridge uplift is represented by an anticline that plunges eastward into the Pasco Basin (Tolan and Reidel 1989). The eastern extension of Yakima Ridge is mostly buried beneath late Cenozoic sediments and has much less structural relief than the rest of Yakima Ridge.

The 200 and 300 Areas are situated on the south flank of the Umtanum-Gable Mountain anticline where the Miocene-aged basalt bedrock dips to the southwest into the Cold Creek syncline. The 100 Areas lie north of the Umtanum Ridge-Gable Mountain anticline in the Wahluke syncline. The deepest parts of the Cold Creek syncline the Wye Barricade depression...
and the Cold Creek depression are approximately 7.5 mi southeast of the 200 Areas and under the 200 West Area respectively.

1435 Geology of the 100 K Area The 100 K Area and vicinity is underlain by the CRBG and intercalated Ellensburg Formation, the Ringold Formation, the Hanford formation, and Holocene deposits. The 100 K Area is near the axis of the Wahluke syncline. See Section A1 4 3 in Annex A for discussion of 200 East Area geology that relates to the CSB.

1436 Tectonic Development of the Hanford Site A seismic hazards analysis of the Hanford Site (WHC-SD-W236A-T1-002) has shown that the geologic history of the Hanford Site from the Precambrian to the present and the resulting geologic structures significantly impact the hazards analysis. This section summarizes the principal geologic events in the development of the Site geology.

The present structure of the Columbia Basin is the product of north-south compression that began in the early Tertiary before the eruption of the CRBG and continues today. The Columbia Basin is composed of two fundamental subprovinces, the Palouse Slope and the YFB. The Palouse Slope is a stable, undeformed area overlying the old continental craton. The YFB overlies a large pre-basalt basin that has been subsiding since the early Tertiary. The pre-CRBG sedimentary basin and CRBG basin are divided by the HR-NR anticline. The edge of the old continental craton lies at the junction of the two structural subprovinces and is presently marked by the Ice Harbor dike swarm of the CRBG.

The pattern of deformation in the Columbia Basin has been dominated by north-south compression and subsidence. The YFB is the principal product of these. Deformation has controlled the location of the Columbia River system since the late Miocene as well as the depositional pattern of the post-basalt sediments. The rates of deformation in the Columbia Basin have declined since the early Tertiary. The rate of regional subsidence of the basin and the rate of local uplift on the anticlinal ridges have both declined. The present rate of ridge growth is estimated at 0.0016 in/yr and the rate of subsidence in the basin is estimated at 0.0001 in/yr.

Microseismicity, high in situ stress conditions, and the geometry of Quaternary-Holocene faulting indicate that the basin is still experiencing north-south compression. Although known late Cenozoic faults are found exclusively on the anticlinal ridges, earthquake focal mechanisms and strain measurements suggest that most stress release is occurring in the synclinal areas. No earthquake events have been shown to be related to known faults. The high in situ stress in the Cold Creek syncline explains the microseismicity in the region, but the absence of microseismicity associated with the anticlinal ridges may result from weakened fault zones lubricated with groundwater that have a component of a seismic or below-detection-limit seismic slip or the fault zones may be locked up.

14361 The Pre-Miocene Columbia Basin The beginning of the development of the Pacific Northwest can be traced to the late Precambrian rifting of the Proterozoic supercontinent when plates that were to become part of either Siberia or Australia separated from North America. The western edge of the North American craton in Washington was near the present...
119° longitude trending north-south in the vicinity of the present Ice Harbor Dam and near the eastern border of the Hanford Site. Between the initial rifting and early Mesozoic time, eastern Washington was a passive margin accumulating sediments along the coast through most of the Paleozoic and early Mesozoic.

When Pangea began to separate in the Triassic, the western coast of North America became an active continental margin with a subduction zone forming along the cratonic boundary. By the late Jurassic, microcontinents and exotic terranes that formed at great distances from North America began to be accreted onto the old craton, expanding the North American continent westward. Washington and the west coast of North America grew by accreting other terranes until at least the Eocene. Presently, these accreted terranes form the crystalline basement of the Hanford Site.

The process of subduction and accretion resulted in suturing of these exotic terranes by the intrusion of batholiths beginning in the Cretaceous and continuing into the Eocene. At the same time, most of eastern Washington was uplifted. Streams and rivers eroded these uplands and deposited the sediments along the edge of the continental shelf. The accretion also resulted in successive westward repositionings of the subduction zone until the present position was reached.

The Cenozoic saw the initiation of extensive volcanism throughout the Pacific Northwest along north-south trending volcanic arcs and in areas where extension occurred during the Eocene. In the Columbia Basin, the Eocene and early Oligocene saw rapid subsidence caused by extension accompanied by infilling of continental sediments and volcanic rocks.

Before the eruption of the CRBG in Miocene time, the Columbia Basin had two distinct parts. The eastern basin was underlain by ancient cratonic rocks that formed a stable gently dipping slope to the west. The western portion of the Columbia Basin was an area underlain by the accreted terranes of the Mesozoic which were overlain by continental volcanic rocks and continental sediments that accumulated in a rapidly subsiding basin during the early part of the Cenozoic. The Hanford Site is along the boundary of these two parts of the Columbia Basin. In "Crustal Structure of the Columbia Plateau – Evidence for Continental Rifting" (Catchings and Mooney 1988), Catchings and Mooney interpreted the basin underlying the western part as a failed rift based on seismic refraction survey. Other studies (RHO-BW-SA-289P, RHO-BW-SA-435P), however, question this interpretation. These studies did not find evidence for a rift basin in the deep crust or mantle.

Tertiary sedimentary and volcanioclastic rocks extend beyond the margin of the CRBG and form part of the Cascade Range (Reidel et al. 1994). In addition, there is no known fault zone along the western margin of the basin as suggested in the rift model. The pre-CRBG sediments that continue across the present Cascade Range were arched upward during intrusion and uplift of the Cascade Range beginning about 12 Ma to form the present western edge of the Columbia Basin. This supports the previously mentioned geophysical studies and suggests that the Columbia Basin is not a rift basin but perhaps simply a back-arc basin. The concept of a rift basin has a significant impact on the seismic hazards of the Site (WHC-SD-W236A-TI-002).
14362 Miocene Columbia Basin  Beginning about 17 Ma and continuing until about 6 Ma, flood-basalt flows of the CRBG began erupting from linear vent systems in eastern Washington, northeast Oregon, and western Idaho. The greatest volume (85%) of basalt was erupted between 16.5 and 15.5 Ma with waning eruptions until 6 Ma. The basalt eruptions were huge by comparison to eruptions today, individual flows contained as much as 1,200 m$^3$ of basalt (Reidel et al. 1989b). The average hiatus between eruptions was as long as 20,000 years.

Borehole, geophysical, and stratigraphic data (Catchings and Mooney 1988; Reidel et al. 1989b) indicate that the CRBG is thinner on the Palouse Slope than in the YFB. The CRBG ranges from 1,640 to 4,920 ft thick on the Palouse Slope but abruptly thickens to as much as 13,125 ft in the Pasco Basin area under the Hanford Site (Reidel et al. 1989b). Regional thickness patterns for both the CRBG and underlying Tertiary sediments indicate that the pre-CRBG basin where the Hanford Site is located began subsiding again. By far the most significant tectonic activity was continued subsidence in the basin. The subaerial nature of the CRBG indicates that subsidence continued as long as basalt was being erupted and that basalt accumulation kept pace with subsidence (Reidel et al. 1989a and 1989b). Subsidence rates from 17 to 15.6 Ma were approximately 0.4 in/yr initially and decreased to 0.001 in/yr in the late Miocene (Reidel et al. 1989b).

During the eruption of the CRBG, the anticlinal ridges were topographic highs against which the basalt flows thinned as they were emplaced (Reidel 1984; Reidel et al. 1989b). Detailed analysis of borehole and field data (Reidel 1984, Reidel et al. 1989b) established a quantitative relationship between flow thickness and fold growth rates. During the initial eruption of the CRBG (17 to 15.6 Ma), the ridges grew at about 0.01 in/yr, and the rate decreased to about 0.002 in/yr during the waning phases (15.6 to 10.5 Ma) (Reidel 1984, Reidel et al. 1989b).

By the end of the massive eruptions of the CRBG (8.5 Ma) most of the Columbia Basin was a shallow, bowl-shaped nearly featureless plain. The massive eruptions had buried most of the structural and topographic relief in the western part of the Columbia Basin, only the anticlinal ridges that were not buried by younger flows stood above the plain. Across this plain flowed the ancestral Columbia River and its main tributaries including the Salmon-Clearwater, Yakima, and Palouse Rivers.

14363 The Late-Miocene to Middle-Pliocene Columbia Basin  The post-CRBG tectonic history of the Columbia Basin and Hanford Site is recorded in the Yakima folds and post-CRBG sediments. Alluvial-lacustrine sediments deposited primarily by the Columbia River system show that the Yakima folds were growing and displacing river channels during the late Miocene and Pliocene (Fecht et al. 1987).

Progressive changes in the distribution of post-CRBG sedimentary facies provide one of the best records of the post-CRBG tectonic history of the Columbia Basin and Hanford Site. Ridge uplift and basin subsidence are recorded by progressive lateral shifts in these depositional environments over time (Fecht et al. 1987, Smith 1988, WHC-SD-EN-TI-008, Reidel et al. 1994).
During the waning phases of CRBG eruptions (12.5 to 8.5 Ma) the Columbia River flowed south across the YFB. Before approximately 8 Ma, the post-CRBG pre-Ringold channel of the Columbia River flowed across the western Pasco Basin, entering at Sentinel Gap and exiting near the southwestern side of Hanford, from there it flowed southwest toward Goldendale. About 8 Ma the Columbia River began to shift eastward onto the Hanford Site flowing south through Gable Gap and occupying a water gap over the eastern end of Rattlesnake Mountain near Benton City, Washington (Fecht et al. 1987). By middle Ringold time (approximately 6 Ma), the Columbia River shifted position again, exiting the Pasco Basin at Wallula Gap as it does now (Fecht et al. 1987).

About 3.4 Ma, regional uplift in western North America caused incision of the Columbia River system that resulted in the removal of over 328 ft of Ringold section across the Hanford Site. This regional incision and erosion produced the White Bluffs along the Columbia River on the eastern boundary of the Hanford Site.

14364 Quaternary Deformation in the Columbia Basin. Uplifted and faulted Ringold and coeval sediments flank most ridges in the central Columbia Basin (Reidel 1984, Reidel et al. 1989b). Deformed Pliocene-Pleistocene sediments also are found on many ridges. Younger glaciofluvial sediments of the Hanford formation locally record some of the youngest deformation in the Columbia Basin.

Evidence for continued growth of the YFB in the Quaternary is mainly restricted to the frontal fault zones. Although not common, evidence of young faulting or suspected young faulting has been found at many locations across the YFB (see Tables 1-19 and 1-20). Young faults have been described at Toppenish Ridge, at Union Gap in Ahtanum Ridge, on Gable Mountain along Umtanum Ridge, in the Columbia Hills anticline, and along the CLEW. Although age relationships are not fully understood, they suggest that faulting has continued since the last catastrophic flood (approximately 13,000 years before present).

1437 Contemporary Stress and Strain

14371 Seismicity. Seismic monitoring at Hanford began when the U.S. Geological Survey installed a small array of seismograph stations around the Hanford Site in the summer of 1969. In 1982 a closely spaced seismic network was installed at the Hanford Site to characterize the microseismicity on the Site for a possible high-level waste repository. This operated until 1988. In 1988 the number of stations in the network was reduced. Earthquakes of magnitudes 1.0 (Coda Amplitude Magnitude) and larger currently can be detected and located at the Hanford Site and earthquakes of magnitude 2.5 and larger are detected and located throughout most of eastern Washington.

Past seismic hazard studies at the Hanford Site have shown that earthquakes can be related to three crustal layers and five general sources (Tables 1-21 and 1-22). All layers and sources are monitored at the Site except the Cascadia Subduction Zone, which is monitored by the University of Washington.
<table>
<thead>
<tr>
<th>Fault</th>
<th>Primary structural feature</th>
<th>Age of last movement</th>
<th>Sense of movement</th>
<th>Amount of movement</th>
<th>Location</th>
</tr>
</thead>
<tbody>
<tr>
<td>Central Ferry</td>
<td>Palouse Slope</td>
<td>Pleistocene(^9)</td>
<td>Sinistral oblique slip</td>
<td>1 to 1.5 m</td>
<td>Center sec 22 T12N R40E</td>
</tr>
<tr>
<td>Thorn Hollow</td>
<td>Hite Fault System</td>
<td>Early Holocene(^9)</td>
<td>Strike-slip</td>
<td>Not determined</td>
<td>SW(^\frac{1}{4})NE(^\frac{1}{4}) sec 2 T4N R35E</td>
</tr>
<tr>
<td>Buroker</td>
<td>Wallula Fault System</td>
<td>Pleistocene to early Holocene(^9)</td>
<td>Thrust fault with component of sinistral strike slip</td>
<td>Greater than 1 m</td>
<td>Sec 31 T7N R37E</td>
</tr>
<tr>
<td>Little Dry Creek</td>
<td>Wallula Fault System</td>
<td>Pleistocene(^9)</td>
<td>Normal</td>
<td>0.5 m</td>
<td>NE(^\frac{1}{4}) sec 11 T4N R35E</td>
</tr>
<tr>
<td>Barrett</td>
<td>Wallula Fault System</td>
<td>Late Pleistocene to Holocene(^9)</td>
<td>Dextral oblique slip</td>
<td>Varied 2 to 50 cm</td>
<td>SW(^\frac{1}{4})SE(^\frac{1}{4}) sec 25 T6N R34E</td>
</tr>
<tr>
<td>Milton Freewater</td>
<td>Wallula Fault System</td>
<td>Holocene (1936)</td>
<td>Dextral strike slip</td>
<td>Unknown ground disturbances</td>
<td>SE(^\frac{1}{4}) sec 18 T5N R35E</td>
</tr>
<tr>
<td>Promontory Point</td>
<td>Wallula Fault System</td>
<td>Pleistocene(^9)</td>
<td>Normal</td>
<td>NK</td>
<td>Sec 10 T6N R37E</td>
</tr>
<tr>
<td>Wallula (near Warm Springs Canyon)</td>
<td>Wallula Fault System</td>
<td>Pleistocene(^9)</td>
<td>Strike slip or oblique slip</td>
<td>NK</td>
<td>Sec 12 T6N R32E</td>
</tr>
<tr>
<td>Athena</td>
<td>Wallula Fault System</td>
<td>Pleistocene(^9)</td>
<td>Oblique slip</td>
<td>NK</td>
<td>Sec 15 T5N R34E</td>
</tr>
<tr>
<td>Wallula (near Vansycle Canyon)</td>
<td>Wallula Fault System</td>
<td>Early Holocene</td>
<td>Not determined</td>
<td>NK</td>
<td>Sec 3 T6N R32E</td>
</tr>
<tr>
<td>Finley Quarry</td>
<td>Cle Elum Wallula Imaent Domain II</td>
<td>Pleistocene(^9)</td>
<td>Reverse</td>
<td>NK</td>
<td>Sec 3 T7N R30E</td>
</tr>
<tr>
<td>Central Gable Mountain</td>
<td>Gable Mountain Anticline (Yakima fold)</td>
<td>Late Pleistocene(^9)</td>
<td>Reverse</td>
<td>5 cm</td>
<td>Sec 19 T13N R27E</td>
</tr>
<tr>
<td>Mill Creek thrust fault and numerous unnamed faults</td>
<td>Toppenish Ridge (Yakima fold)</td>
<td>Holocene</td>
<td>Both normal and reverse</td>
<td>Up to 4 m</td>
<td>Area between latitude 46 15' -46 19' longitude 120 22' - 122 40'</td>
</tr>
<tr>
<td>Union Gap</td>
<td>Ahtanum Ridge (Yakima fold)</td>
<td>Pleistocene(^9)</td>
<td>Reverse</td>
<td>Approximately 7 m</td>
<td>T12N R19E</td>
</tr>
<tr>
<td>Smyrna Bench</td>
<td>Saddle Mountains (Yakima Fold)</td>
<td>Late Pleistocene Holocene(^9)</td>
<td>Normal reverse</td>
<td>6 m(^3)</td>
<td>T15 16N R25 27E</td>
</tr>
</tbody>
</table>

\(^{9}\) = not known  
R = range  
T = township
### Table 1-20 Earthquakes Equal to or Greater than Modified Mercalli Intensity V in the Columbia Plateau and Surrounding Area from 1870 through 1980

<table>
<thead>
<tr>
<th>Date</th>
<th>Universal time</th>
<th>Epicentral intensity magnitude</th>
<th>Coordinates</th>
<th>Location/remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>March 5 1892</td>
<td>LT</td>
<td>VI</td>
<td>46.6 N 120.5°W</td>
<td>North Yakima Washington</td>
</tr>
<tr>
<td>March 5 1893</td>
<td>LT</td>
<td>VI</td>
<td>45.9 N 119.3°W</td>
<td>Umatilla Oregon</td>
</tr>
<tr>
<td>July 5 1911</td>
<td>08 00</td>
<td>V</td>
<td>47.0 N 120.5°W</td>
<td>Ellensburg Washington</td>
</tr>
<tr>
<td>February 28 1918</td>
<td>23 15</td>
<td>V</td>
<td>46.5 N 120.5°W</td>
<td>Yakima Washington</td>
</tr>
<tr>
<td>November 1 1918</td>
<td>17 20</td>
<td>VI</td>
<td>46.7°N 119.5°W</td>
<td>Corfu Washington</td>
</tr>
<tr>
<td>September 14 1921</td>
<td>11 00</td>
<td>VI</td>
<td>46.1 N 118.25°W</td>
<td>Dixie Walla Walla Washington</td>
</tr>
<tr>
<td>September 18 1934</td>
<td>24 00 LT</td>
<td>V</td>
<td>47.0°N 120.5°W</td>
<td>Ellensburg Washington</td>
</tr>
<tr>
<td>September 26 1934</td>
<td>16 15 LT</td>
<td>V</td>
<td>47.0 N 120.5°W</td>
<td>Ellensburg Washington</td>
</tr>
<tr>
<td>September 26 1934</td>
<td>16 45</td>
<td>V</td>
<td>47.0 N 120.5°W</td>
<td>Ellensburg Washington</td>
</tr>
<tr>
<td>September 26 1934</td>
<td>21 15</td>
<td>V</td>
<td>47.0 N 120.5°W</td>
<td>Ellensburg Washington</td>
</tr>
<tr>
<td>October 19 1934</td>
<td>23 31 LT</td>
<td>V</td>
<td>47.0°N 120.5°W</td>
<td>Ellensburg Washington</td>
</tr>
<tr>
<td>November 1 1934</td>
<td>07 28</td>
<td>V</td>
<td>47.0 N 120.5°W</td>
<td>Ellensburg Washington</td>
</tr>
<tr>
<td>November 2 1934</td>
<td>15 17 LT</td>
<td>V</td>
<td>47.0 N 120.5°W</td>
<td>Ellensburg Washington</td>
</tr>
<tr>
<td>July 16 1936</td>
<td>07 07 49 0</td>
<td>VII</td>
<td>46.2 N 118.20°W</td>
<td>Milton Freewater Oregon (WCC Relocated)</td>
</tr>
<tr>
<td>August 4 1936</td>
<td>09 19</td>
<td>V</td>
<td>45.8 N 118.6°W</td>
<td>Helix Oregon</td>
</tr>
<tr>
<td>August 28 1936</td>
<td>04 39</td>
<td>V</td>
<td>45.9 N 118.4°W</td>
<td>Milton Freewater Oregon</td>
</tr>
<tr>
<td>October 31 1944</td>
<td>11 34 28 7</td>
<td>V</td>
<td>47.8 N 120.6°W</td>
<td>Fish Lake Washington</td>
</tr>
<tr>
<td>January 13 1948</td>
<td>06 55 00</td>
<td>V</td>
<td>47.9 N 120.3°W</td>
<td>Lucerne Waterville Washington</td>
</tr>
<tr>
<td>January 7 1951</td>
<td>22 45 00</td>
<td>V</td>
<td>45.9 N 119.2°W</td>
<td>McNary Oregon</td>
</tr>
<tr>
<td>January 20 1959</td>
<td>About 23 15</td>
<td>V</td>
<td>46.2 N 118.2°W</td>
<td>Milton Freewater Oregon</td>
</tr>
<tr>
<td>July 23 1966</td>
<td>01 57 08 8</td>
<td>4 3 MB</td>
<td>47.2 N 119.5°W</td>
<td>Ephrata Washington</td>
</tr>
<tr>
<td>December 20 1973</td>
<td>01 08 28 2</td>
<td>V 4 4 MC</td>
<td>46.9 N 119.35°W</td>
<td>Corfu Washington 24 km depth</td>
</tr>
<tr>
<td>April 8 1979</td>
<td>07 29 37 8</td>
<td>4 2 MC</td>
<td>46.0 N 118.4°W</td>
<td>Walla Walla Washington (UW)</td>
</tr>
</tbody>
</table>


Latitude and longitude are used to define the location of historical earthquakes. Some times and coordinates have been modified from the original source times and coordinates to better reflect the possible error of these early earthquakes.

- **LT** = local time
- **MB** = body wave magnitude
- **MC** = coda length magnitude
- **ML** = local magnitude
- **MS** = surface wave magnitude
- **UW** = University of Washington
- **WCC** = Woodward Clyde Consultants
Table 1-21 Depth of Earthquakes in the Columbia Basin

<table>
<thead>
<tr>
<th>Layer</th>
<th>Depth</th>
</tr>
</thead>
<tbody>
<tr>
<td>Columbia River Basalt Group</td>
<td>0-5 km</td>
</tr>
<tr>
<td>Prebasalt sediments</td>
<td>5-10 km</td>
</tr>
<tr>
<td>Crystalline basement</td>
<td>&gt;10 km</td>
</tr>
</tbody>
</table>

Table 1-22 Principal Locations of Earthquakes in the Columbia Basin

<table>
<thead>
<tr>
<th>Area</th>
<th>Layer</th>
</tr>
</thead>
<tbody>
<tr>
<td>Major reverse faults on ridges</td>
<td>Mainly Columbia River Basalt Group, also prebasalt sediments</td>
</tr>
<tr>
<td>Secondary faults on ridges</td>
<td>Columbia River Basalt Group</td>
</tr>
<tr>
<td>Swarm area</td>
<td>Columbia River Basalt Group</td>
</tr>
<tr>
<td>Basement</td>
<td>Crystalline basement</td>
</tr>
<tr>
<td>Cascadia Subduction Zone</td>
<td>Lithosphere - plate tectonic boundary, outside the Columbia Basin</td>
</tr>
</tbody>
</table>

**Vertical Patterns** There are three horizontal layers of seismicity (seismic stratigraphy) related to the stratigraphy of the Hanford Site and vicinity (Table 1-21) the CRBG the prebasalt sediments, and the crystalline basement. About 75% of the earthquakes have originated in the CRBG layer. The prebasalt sedimentary layer has had 8% of the events and the crystalline basement has had 17%.

**Shallow Earthquakes in the Basalt** The majority of the seismicity at the Hanford Site and the surrounding area comes from the basalt layer that extends from the surface to approximately 2.5 m under the Site.

**Earthquakes in Sedimentary Rock below the Basalt** The seismicity in the pre-CRBD sedimentary rock appears to be confined to the top 1.9 m. The seismicity of this sedimentary layer at the Hanford Site is relatively low when compared to the basalt layer but may be related to localized detachment zones related to the growth of the anticlinal structures.

**Earthquakes in the Crystalline Basement** Deep earthquakes below 6.2 m appear to be concentrated in the western and southwestern portions of the Hanford Site. The deepest earthquakes located below the Site are shallower than 18.6 m.
Using first-motion data from the Eastern Washington Regional Network and from the Basalt Waste Isolation Project, focal mechanisms show faulting that strikes between N 30° W and N 80° W. Although the strike is consistently west-northwest, the throw on the assumed faults varies. These data indicate reverse faults or strike-slip faults.

**Spatial Patterns** Past studies at the Hanford Site have concluded that there are five different tectonic environments (earthquake sources) where earthquakes can occur near the Hanford Site and in the Columbia Basin of eastern Washington (Table 1-22):

1. Reverse/thrust faults in the CRBG associated with major anticlinal ridges such as Rattlesnake Mountain, Yakima Ridge, and Umtanum Ridge

2. Secondary faults occurring on the major anticlinal ridges

3. Small geographic areas of unknown geologic structure that produce clusters of events (swarms) usually in the CRBG in synclinal valleys

4. Basement source structures, although earthquakes cannot be directly tied to a mapped fault because very little is known about geologic structures in the crystalline basement beneath the Hanford Site

5. The Cascadia Subduction Zone, which recently has been postulated to be capable of producing a magnitude 9 earthquake.

**Floating Earthquakes** A "floating" earthquake within the tectonic environment covering the entire Columbia Basin, including the Hanford Site, has been considered. A floating earthquake is one that, for seismic design purposes, can happen anywhere in a tectonic province and is not associated with any known geologic structure. It can be floated anywhere in the province.

**Earthquake Swarm Areas** The major source of earthquakes at the Hanford Site is swarm activity in the syncline of the YFB. There are three general areas of significant swarm activity: the Wooded Island swarm area, Coyote Rapids swarm area, and the Saddle Mountains swarm area (Figure 1-26).

The Wooded Island swarm area, located near the 300 Area, occurs at the eastern edge of the YFB where it abuts against the Palouse Slope. This boundary marks the suture zone between the old accreted terranes to the west and the stable Precambrian-Paleozoic craton to the east (Reidel et al. 1994). This zone is marked by an abrupt increase in the thickness of the basalt and subbasalt sediment over the accreted terranes and abrupt thinning of the basalt and sediment over the craton.

The Coyote Rapids Swarm Area is located at the horn of the Columbia between 100 K and 100 N areas. It occurs over no known geologic structure. The K Basins and CVDF are located in the southern portion of this swarm. The swarm lies at the intersection of two paleoslopes that make a northeast-southwest trough extending from Spokane, Washington, to the Columbia Gorge. This zone may be an old basement weakness zone, but there is no known reason for the
swarm to occur in its present position. The largest earthquake recorded on the Hanford Site has a magnitude of 3.8 and was a part of the Coyote Rapids Swarm Area.

The Saddle Mountains Swarm Area is located along the north side of the Saddle Mountains. The swarm area is located north of the Saddle Mountains fault zone in an area that has no mapped geologic structures. There is evidence for recent (post-13,000 years) faulting on the Saddle Mountain fault (Table 1-19), but there is no evidence for faults at the swarm. The cause of the earthquake swarm is not known at this time.

**Magnitude of Earthquakes** Earthquake activity at Hanford and in the Columbia Basin is summarized in Tables 1-23, 1-24, and 1-25. Since July 1982, approximately 650 earthquakes between magnitude 0 and 3.8 have been recorded on and around the Hanford Site. The pattern of earthquake activity between 1990 and 1995 is shown in Figure 1-27. The greatest seismic activity is in the low-magnitude range, with only 5 events exceeding a magnitude of 3.0. The largest magnitude earthquake on the Hanford Site was a 3.8 magnitude earthquake on October 25, 1971, in the Coyote Rapids Swarm Area (Figure 1-28) (Table 1-25). The largest recent, felt earthquake was a 3.3 magnitude earthquake on June 12, 1995, in the Wooded Island swarm area. The largest regional earthquake was the 5.7 Milton-Freewater earthquake on July 16, 1936 (Table 1-24). This earthquake occurred 62 mi southeast of the Hanford Site. The 1936 Milton-Freewater earthquake was estimated to have a peak acceleration of 0.03 g at the Hanford Site.

**14372 Contemporary Stress in the Cold Creek Syncline** Contemporary stress in the Cold Creek syncline is expressed principally as horizontal shortening and subsidence. Geodetic surveys (Open File Report 84-797) were performed across the Pasco Basin to determine rates of shortening. The data suggest north-south shortening. The rate of shortening is not statistically significant at the 95% confidence level, however, and the measurements are within the error limits of the recording instruments.

Contemporary stress measurements were performed at the Hanford Site as part of the Basalt Waste Isolation Project. Core disking and spalling in boreholes drilled in the Cold Creek syncline indicate relatively high in situ stress (DOE/RW-0164). Hydraulic fracturing tests were conducted in boreholes in the Cold Creek syncline at a depth of about 0.6 mi (DOE/RW-0164). The results also indicated high in situ stress. The maximum horizontal stress ranges from 7,630 to 9,780 lbf/in² and the minimum horizontal stress ranges from 4,400 to 5,180 lbf/in², with a mean horizontal to vertical ratio of 1.77 ± 0.20. The mean orientation of induced fractures and the direction of the maximum horizontal stress is consistent with north-south compression (DOE/RW-0164).
### Table 1-23  Earthquakes with Magnitude Greater than or Equal to 3 or Intensity Greater than or Equal to IV Occurring in the Columbia Plateau and Surrounding Area from 1866 to 1966  (5 sheets)

<table>
<thead>
<tr>
<th>Date</th>
<th>Universal time</th>
<th>Intensity/magnitude</th>
<th>Coordinates</th>
<th>Locations</th>
</tr>
</thead>
<tbody>
<tr>
<td>November 24 1866</td>
<td>18 10</td>
<td>IV</td>
<td>45 6 N 121 2 W</td>
<td>The Dalles Oregon</td>
</tr>
<tr>
<td>December 15 1872</td>
<td>05 40</td>
<td>VIII</td>
<td>49 0 N 121 0 W</td>
<td>Lake Chelan Washington</td>
</tr>
<tr>
<td>September 2 1891</td>
<td>10 30 LT</td>
<td>IV</td>
<td>47 1 N 118 4 W</td>
<td>Ritzville Washington</td>
</tr>
<tr>
<td>September 17 1891</td>
<td>04 30</td>
<td>IV</td>
<td>44 9 N 121 0 W</td>
<td>Salem Oregon</td>
</tr>
<tr>
<td>February 29 1892</td>
<td>10 45</td>
<td>IV</td>
<td>45 6 N 121 2 W</td>
<td>The Dalles Oregon</td>
</tr>
<tr>
<td>March 5 1892</td>
<td>LT</td>
<td>VI</td>
<td>46 6 N 120 5 W</td>
<td>North Yakima Washington</td>
</tr>
<tr>
<td>March 5 1893</td>
<td>LT</td>
<td>VI</td>
<td>45 9 N 119 3 W</td>
<td>Umatilla Oregon</td>
</tr>
<tr>
<td>December 15 1897</td>
<td>LT</td>
<td>V</td>
<td>47 8 N 120 0 W</td>
<td>Lakeside Washington</td>
</tr>
<tr>
<td>October 18 1905</td>
<td>23 LT</td>
<td>V</td>
<td>47 8 N 120 0 W</td>
<td>Chelan Washington</td>
</tr>
<tr>
<td>January 2 1906</td>
<td>LT</td>
<td>VI</td>
<td>48 7 N 117 8 W</td>
<td>Stevens County Washington</td>
</tr>
<tr>
<td>November 2 1906</td>
<td>01 49</td>
<td>V</td>
<td>48 5 N 117 9 W</td>
<td>Reported felt information</td>
</tr>
<tr>
<td>February 18 1907</td>
<td>12 20 LT</td>
<td>V</td>
<td>47 8 N 120 0 W</td>
<td>Chelan Washington</td>
</tr>
<tr>
<td>January 21 1909</td>
<td>05 LT</td>
<td>IV</td>
<td>47 8 N 120 0 W</td>
<td>Chelan Washington</td>
</tr>
<tr>
<td>May 24 1909</td>
<td>22 LT</td>
<td>V</td>
<td>47 7 N 120 4 W</td>
<td>Chelan Leavenworth Washington</td>
</tr>
<tr>
<td>June 12 1908</td>
<td>Unknown</td>
<td>V</td>
<td>45 0 N 117 25 W</td>
<td>Cornucopia Oregon</td>
</tr>
<tr>
<td>July 5 1911</td>
<td>08 00</td>
<td>V</td>
<td>47 0 N 120 5 W</td>
<td>Ellensburg Washington</td>
</tr>
<tr>
<td>October 14 1913</td>
<td>23 00</td>
<td>V</td>
<td>45 7 N 117 1 W</td>
<td>Seven Devils Idaho</td>
</tr>
<tr>
<td>March 5 1915</td>
<td>05 10</td>
<td>IV</td>
<td>47 8 N 120 0 W</td>
<td>Lakeside Washington</td>
</tr>
<tr>
<td>March 5 1915</td>
<td>05 30</td>
<td>IV</td>
<td>47 8 N 120 0 W</td>
<td>Lakeside Washington</td>
</tr>
<tr>
<td>July 18 1915</td>
<td>20 54</td>
<td>IV</td>
<td>47 8 N 120 0 W</td>
<td>Lakeside Washington</td>
</tr>
<tr>
<td>August 18 1915</td>
<td>14 05</td>
<td>V</td>
<td>48 5 N 121 4 W</td>
<td>Felt over 78 000 km² (30 000 mi²)</td>
</tr>
<tr>
<td>December 10 1915</td>
<td>20 45</td>
<td>IV</td>
<td>47 7 N 117 4 W</td>
<td>Spokane Washington</td>
</tr>
<tr>
<td>February 21 1918</td>
<td>LT</td>
<td>IV</td>
<td>46 9 N 121 3 W</td>
<td>Bumping Lake Washington</td>
</tr>
<tr>
<td>February 28 1918</td>
<td>23 15</td>
<td>V</td>
<td>46 5 N 120 5 W</td>
<td>Near Yakima Washington</td>
</tr>
<tr>
<td>March 12 1918</td>
<td>03 26</td>
<td>V</td>
<td>47 6 N 117 0 W</td>
<td>Spokane Washington</td>
</tr>
<tr>
<td>April 18 1918</td>
<td>21 13</td>
<td>IV</td>
<td>47 6 N 117 4 W</td>
<td>White Bluffs Prairie Washington</td>
</tr>
<tr>
<td>November 1 1918</td>
<td>17 20</td>
<td>VI</td>
<td>46 7 N 119 5 W</td>
<td>Corfu Washington</td>
</tr>
<tr>
<td>October 7 1920</td>
<td>02 LT</td>
<td>V</td>
<td>47 6 N 120 1 W</td>
<td>Waterville Washington</td>
</tr>
<tr>
<td>November 28 1920</td>
<td>11 30</td>
<td>IV V</td>
<td>45 7 N 121 5 W</td>
<td>Hood River Oregon</td>
</tr>
</tbody>
</table>
Table 1-23  Earthquakes with Magnitude Greater than or Equal to 3 or Intensity Greater than or Equal to IV Occurring in the Columbia Plateau and Surrounding Area from 1866 to 1966 (5 sheets)

<table>
<thead>
<tr>
<th>Date</th>
<th>Universal time</th>
<th>Intensity*</th>
<th>Coordinates</th>
<th>Locations</th>
</tr>
</thead>
<tbody>
<tr>
<td>September 14, 1921</td>
<td>11 00</td>
<td>VI</td>
<td>46 1 N 118 2 W</td>
<td>Dixie Walla Walla Washington</td>
</tr>
<tr>
<td>June 1, 1922</td>
<td>23 30</td>
<td>IV</td>
<td>47 7 N 117 4 W</td>
<td>Spokane Washington</td>
</tr>
<tr>
<td>January 6, 1924</td>
<td>13 09</td>
<td>IV</td>
<td>46 1 N 118 3 W</td>
<td>Walla Walla Washington</td>
</tr>
<tr>
<td>January 6, 1924</td>
<td>23 10</td>
<td>V</td>
<td>45 8 N 118 3 W</td>
<td>Milton and Weston Oregon</td>
</tr>
<tr>
<td>May 27, 1924</td>
<td>00 19 00</td>
<td>IV</td>
<td>46 1 N 118 3°W</td>
<td>Walla Walla Washington</td>
</tr>
<tr>
<td>November 28, 1925</td>
<td>01 25 00</td>
<td>4.30 ML</td>
<td>47 5 N 116 0 W</td>
<td>--</td>
</tr>
<tr>
<td>April 23, 1926</td>
<td>13 56 00</td>
<td>IV</td>
<td>46 1 N 118 3 W</td>
<td>Walla Walla Washington</td>
</tr>
<tr>
<td>October 17, 1926</td>
<td>02 45 00</td>
<td>V</td>
<td>45 7 N 121 5 W</td>
<td>White Salmon Washington</td>
</tr>
<tr>
<td>November 27, 1926</td>
<td>18 25 LT</td>
<td>V</td>
<td>47 5 N 116 0 W</td>
<td>Near Rathdrum Idaho</td>
</tr>
<tr>
<td>December 30, 1926</td>
<td>17 57 00</td>
<td>VI</td>
<td>47 7 N 120 2 W</td>
<td>Chelan East Central Washington</td>
</tr>
<tr>
<td>January 3, 1927</td>
<td>04 58 00</td>
<td>VI</td>
<td>47 6 N 120 6 W</td>
<td>Leavenworth, Washington</td>
</tr>
<tr>
<td>April 8, 1927</td>
<td>05 00</td>
<td>V</td>
<td>44 8° N 117 2 W</td>
<td>Richland Oregon</td>
</tr>
<tr>
<td>September 3, 1930</td>
<td>13 00 00</td>
<td>V</td>
<td>47 3 N 117 8 W</td>
<td>Near Lamont Washington</td>
</tr>
<tr>
<td>December 8, 1931</td>
<td>14 25 00</td>
<td>IV</td>
<td>47 8 N 120 0 W</td>
<td>Lakeside Chelan Falls, Washington</td>
</tr>
<tr>
<td>May 31, 1933</td>
<td>20 20 00</td>
<td>IV</td>
<td>47 8 N 120 0 W</td>
<td>Chelan, Washington</td>
</tr>
<tr>
<td>May 31, 1933</td>
<td>20 30 00</td>
<td>IV</td>
<td>47 8 N 120 0 W</td>
<td>Chelan, Washington</td>
</tr>
<tr>
<td>March 9, 1934</td>
<td>16 00 00</td>
<td>IV</td>
<td>47 8 N 120 0 W</td>
<td>Lakeside, Washington</td>
</tr>
<tr>
<td>September 18, 1934</td>
<td>24 LT</td>
<td>V</td>
<td>47 0 N 120 5 W</td>
<td>Ellensburg, Washington</td>
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Table 1-23 Earthquakes with Magnitude Greater than or Equal to 3 or Intensity Greater than or Equal to IV Occurring in the Columbia Plateau and Surrounding Area from 1866 to 1966 (5 sheets)

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Table 1-23  Earthquakes with Magnitude Greater than or Equal to 3 or Intensity Greater than or Equal to IV Occurring in the Columbia Plateau and Surrounding Area from 1866 to 1966 (5 sheets)

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Table 1-23  Earthquakes with Magnitude Greater than or Equal to 3 or Intensity Greater than or Equal to IV Occurring in the Columbia Plateau and Surrounding Area from 1866 to 1966 (5 sheets)

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*Modified Mercalli Intensity


Latitude and longitude are used to define the location of historical earthquakes. Some times and coordinates have been modified from the original source times and coordinates to better reflect the possible error of these early earthquakes.

LT = local time
MB = body wave magnitude
ML = local magnitude
MS = surface wave magnitude
### Table 1-24 Catalog for the Hanford Site and Columbia Plateau of Earthquakes with Magnitudes Greater than or Equal to 3 Located by Instruments from 1969 to 1995 (5 sheets)

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## Table 1-24  Catalog for the Hanford Site and Columbia Plateau of Earthquakes with Magnitudes Greater than or Equal to 3 Located by Instruments from 1969 to 1995 (5 sheets)

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**Notes:**
- Magnitude* signifies the magnitude of the earthquake.
- Coordinates are given in degrees, minutes, and seconds.
- Depth is given in kilometers.
Table 1-24  Catalog for the Hanford Site and Columbia Plateau of Earthquakes
with Magnitudes Greater than or Equal to 3 Located by Instruments
from 1969 to 1995  (5 sheets)

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## Table 1-24 Catalog for the Hanford Site and Columbia Plateau of Earthquakes with Magnitudes Greater than or Equal to 3 Located by Instruments from 1969 to 1995 (5 sheets)

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Table 1-24  Catalog for the Hanford Site and Columbia Plateau of Earthquakes with Magnitudes Greater than or Equal to 3 Located by Instruments from 1969 to 1995  (5 sheets)

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</tr>
<tr>
<td>June 18 1994</td>
<td>07 01 07 28</td>
<td>4 3</td>
<td>47 621°N 121 270°W</td>
<td>0 04</td>
</tr>
<tr>
<td>September 10 1994</td>
<td>07 43 11 37</td>
<td>3 9</td>
<td>47 185°N 121 965°W</td>
<td>17 73</td>
</tr>
<tr>
<td>November 13 1994</td>
<td>16 50 47 20</td>
<td>3 3</td>
<td>46 592°N 119 584°W</td>
<td>28 22</td>
</tr>
<tr>
<td>January 13 1995</td>
<td>19 38 23 02</td>
<td>3 2</td>
<td>46 579°N 120 711°W</td>
<td>13 17</td>
</tr>
<tr>
<td>March 9 1995</td>
<td>07 22 36 62</td>
<td>3 0</td>
<td>47 191°N 120 955°W</td>
<td>1 61</td>
</tr>
<tr>
<td>May 20 1995</td>
<td>12 48 48 20</td>
<td>4 1</td>
<td>46 881°N 121 941°W</td>
<td>13 42</td>
</tr>
<tr>
<td>June 12 1995</td>
<td>01 48 24 40</td>
<td>3 3</td>
<td>46 405°N 119 263°W</td>
<td>1 05</td>
</tr>
<tr>
<td>June 30 1995</td>
<td>22 17 05 06</td>
<td>3 0</td>
<td>47 107°N 120 528°W</td>
<td>11 23</td>
</tr>
<tr>
<td>July 13 1995</td>
<td>10 28 50 27</td>
<td>3 7</td>
<td>46 819°N 121 878°W</td>
<td>8 29</td>
</tr>
<tr>
<td>August 29 1995</td>
<td>13 02 48 76</td>
<td>3 1</td>
<td>46 208°N 119 576°W</td>
<td>15 34</td>
</tr>
<tr>
<td>November 2 1995</td>
<td>14 30 14 44</td>
<td>3 62</td>
<td>46 150°N 119 579°W</td>
<td>23 31</td>
</tr>
</tbody>
</table>

*Unless otherwise noted, all magnitudes are coda length magnitudes.

Note  Latitude and longitude are used to define the location of historical earthquakes.

MB = body wave magnitude
ML = local (Richter) magnitude
NOAA = National Oceanic and Atmospheric Administration
Table 1-25  Swarm Activity at the Hanford Site and Central Columbia Plateau from 1969 through 1995

<table>
<thead>
<tr>
<th>Swarm</th>
<th>Latitude north</th>
<th>Longitude west</th>
<th>Area (km²)</th>
<th>Years of swarms*</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>South</td>
<td>North</td>
<td>East</td>
<td>West</td>
</tr>
<tr>
<td>West Saddle Mountains</td>
<td>46 820</td>
<td>46 865</td>
<td>119 564</td>
<td>119 797</td>
</tr>
<tr>
<td>Frenchman Hills</td>
<td>46 865</td>
<td>46 960</td>
<td>119 514</td>
<td>119 600</td>
</tr>
<tr>
<td>Smyrna</td>
<td>46 800</td>
<td>46 855</td>
<td>119 433</td>
<td>119 564</td>
</tr>
<tr>
<td>Royal</td>
<td>46 850</td>
<td>46 910</td>
<td>119 317</td>
<td>119 450</td>
</tr>
<tr>
<td>Corfu</td>
<td>46 790</td>
<td>46 850</td>
<td>119 331</td>
<td>119 433</td>
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<tr>
<td>Wahluke</td>
<td>46 729</td>
<td>46 780</td>
<td>119 314</td>
<td>119 431</td>
</tr>
<tr>
<td>Othello</td>
<td>46 650</td>
<td>46 710</td>
<td>119 183</td>
<td>119 317</td>
</tr>
<tr>
<td>Scootney Reservoir</td>
<td>46 600</td>
<td>46 680</td>
<td>119 047</td>
<td>119 183</td>
</tr>
<tr>
<td>Connell</td>
<td>46 650</td>
<td>46 715</td>
<td>118 850</td>
<td>118 933</td>
</tr>
<tr>
<td>Wooded Island/Johnson Island</td>
<td>46 390</td>
<td>46 470</td>
<td>119 200</td>
<td>119 333</td>
</tr>
<tr>
<td>Eltopia</td>
<td>46 370</td>
<td>46 455</td>
<td>118 964</td>
<td>119 067</td>
</tr>
<tr>
<td>Coyote Rapids</td>
<td>46 635</td>
<td>46 730</td>
<td>119 483</td>
<td>119 650</td>
</tr>
<tr>
<td>Cold Creek</td>
<td>46 470</td>
<td>46 570</td>
<td>119 60</td>
<td>119 68</td>
</tr>
</tbody>
</table>

*Reflects the total period of recorded swarm activity and may include more than one swarm period

1437 Subsidence  Subsidence has been continuing in the Pasco Basin and Hanford Site since at least the Eocene. Estimates of subsidence since the middle Miocene (17 Ma) indicate the rates of subsidence have declined. From 17 to about 15.6 Ma, the rate of subsidence was about 0.3 to 0.4 m/yr. This decreased to about 0.01 m/yr from 15.6 to 14.5 Ma. The present rate of ridge growth is estimated at 0.0015 m/yr and the rate of subsidence in the basin is estimated at 0.0001 m/yr.

The pattern of subsidence in the Cold Creek syncline is shown in Figure 1-24. There are two broad areas of subsidence in the Cold Creek syncline: the Wye Barricade depression and the Cold Creek Depression. Both areas have had a long slow history of subsidence (Reidel et al. 1989b). Both areas are away from the 200 Areas and the CSB site.

1438 Geologic Hazards  The geologic hazards that affect the performance of the SNF Project facilities have been assessed based on the geologic data addressed in previous sections. These hazards are discussed below and where appropriate quantified for use in the structural evaluations and safety analyses. Geologic hazard assessments for the CVDF and the CSB are found in Section 1438 of the facility FSAR Annexes.
Seismic Hazard Assessment

A seismic hazard analysis has been completed for the Hanford Site (WHC-SD-W236A-TI-002). The Washington Public Power Supply System conducted previous seismic hazard analyses for WNP-1, WNP-4, and WNP-2, which also are located on the Hanford Site (Power et al. 1981). The study was later applied to the DOE-held areas within the Hanford Site by Woodward-Clyde Consultants (WHC-MR-0023). The following discussion is based on WHC-SD-W236A-TI-002 Proportional Seismic Hazard Analysis DOE Hanford Site Washington, the most recent seismic hazard analysis. It incorporates seismo-tectonic data and interpretations that postdate the Washington Public Power Supply System's earlier assessment and are discussed in previous sections of this report. The details of the source models and attenuation relationships used in the hazard assessment are included in WHC-SD-W236A-TI-002.

The potential seismic sources determined to be the major contributors to the seismic hazard in and around the Hanford Site are crustal sources and Cascadia Subduction Zone earthquakes in western Washington State. The crustal sources are:

- Fault sources related to the Yakima folds
- Shallow basalt sources that account for the observed seismicity within the CRBG and not associated with the anticlines
- Crystalline basement source

The site response characteristics of the soils underlying the Hanford Site are similar to those represented in the empirical strong motion database from California. This was determined by comparing the relative response of characteristic Hanford Site soil profiles and dynamic soil properties with those of California deep-soil strong-motion recording stations. Time histories representative of the events contributing to the Hanford Site hazard were used for ground motion input (WHC-SD-W236A-TI-002, Appendix A).

The mean seismic hazard curves for the 100 K Area are shown in Figures 1-29 and 1-30 and illustrate the contributions of individual folds to the hazard. The relative contribution of crustal and Cascadia Subduction sources at the 100 K Area is illustrated in Figure 1-31 and the relative contribution of the three crustal sources for the same location is shown in Figure 1-32.

The performance category 3 horizontal and vertical equal-hazard response spectra were developed for the CVDF site. These are shown at 5% damping for performance category 3 in Figure 1-33. These spectra are the enveloping spectra for the 200 Areas and the 100 K Area. More detail and additional damping values are presented in WHC-SD-W236A-TI-002. The design basis earthquake for the K Basin's safety-class structures is defined in HPS-SDC-4 as an event producing a maximum horizontal ground acceleration of 0.2 g simultaneously with a vertical ground acceleration of 0.13 g.

The remaining 100 K structures were designed and evaluated to Uniform Building Code criteria in effect at the time of their design.
14382 Volcanic Hazard Assessment  Two types of volcanic hazards have affected the Hanford Site in the past 20 million years

- Continental flood basalt volcanism that produced the CRBG, which underlies the Hanford Site, outcropping in the surrounding ridges

- Volcanism associated with the Cascade Range

Several volcanoes in the Cascade Range are currently considered active, but activity associated with flood basalt volcanism has ceased The flood basalt volcanism that produced the CRBG occurred between 17 million and 6 million years before present Most of the lava was extruded during the first 2 to 2.5 million years of the 11-million-year volcanic episode Flood basalt volcanism has not recurred during the last 6 million years, suggesting that the tectonic processes that created the episode have ceased The recurrence of Columbia River basin volcanism is not considered to be a credible volcanic hazard (DOERW-0164)

Volcanism in the Cascade Range has been active throughout the Pleistocene Epoch (approximately 2 million years before present to 10 000 years before present) and through the Holocene Epoch (10 000 years before present to present) The eruption history of the Holocene best characterizes the most likely types of activity in the next 100 years Many of the volcanoes in the Cascade Range have been active in the last 10 000 years including Mount Mazama (Crater Lake) and Mount Hood in Oregon and Mount St Helens, Mount Adams and Mount Rainier in Washington state (Figure 1-23) The Hanford Site is approximately 95 mi from Mount Adams 110 mi from Mount Rainier, and 125 mi from Mount St Helens the three closest active volcanoes At these distances the tephra (ash) is the only hazard Mount St Helens has been considerably more active throughout the Holocene than Mount Rainier or Mount Adams, which is the least active of the three Ashfall at the Hanford Site during the May 18 1980, eruption of Mount St Helens ranged from a trace in the southern part of the Site to about 0.5 in in the 100 Areas

Probabilistic volcanic hazard studies of the Cascade Range have been completed by the U.S. Geological Survey (Open File Report 87-297, Open File Report 95-492) Figure 1-23 illustrates the annual probability of exceeding 0.4 in of volcanic ash accumulation in Washington and Oregon following the eruption of a major Cascade Range volcano and the annual probability of exceeding 4 in of volcanic ash accumulation Figure 1-34 presents this information as a volcanic ash hazard curve for the Hanford Site The design basis ground ash load for safety class SSCs is the same as that for Energy Northwest's WNP-2, 24 lb/ft²

14383 Subsurface Stability  The SNF Project facilities are constructed on flood sediments the youngest sediments being approximately 13 000 years old There are no areas of potential surface or subsurface subsidence, uplift, or collapse except for the low geologic deformation discussed in Section 1436 With the exception of the loose, surficial wind-deposited silt, soils are competent and form good foundations Geotechnical studies have been completed in and around the 100 K Area (WHC-SD-NR-ER-093 Redpath 1994) The water table in the 100 K Area is approximately 75 ft below ground surface in the relatively
compact and moderately well cemented unit of the Rmgold Formation. See Section A1.4.3 in Annex A for the discussion of subsurface stability at the CSB

1.5 NATURAL PHENOMENA THREATS

See Sections 1.5 of the facility FSAR Annexes for a discussion of the natural phenomena design criteria used for their design.

1.6 EXTERNAL HUMAN-GENERATED THREATS

This section identifies and investigates potential human-generated threats to SNF Project operation. The main threats are aircraft crashes and transportation accidents. See Section 1.6 of the facility FSAR Annexes for facility-specific evaluations.

1.6.1 Aircraft Activity

Two airports serving commercial and military aircraft are located within 35 mi of the K Basins and CVDF site and within 30 mi of the CSB site. The closest is the Richland Airport 21 mi south-southeast of the CSB. This airport has two 4,000-ft runways, one with a 010°/190° orientation and the other with a 070°/250° orientation. Runway capability is about 30,000 lb per point of contact. Visual flight rule landings are standard Federal Aviation Administration non-control-tower patterns. Federal arrivals and instrument approaches and departures are shown on Figure 1-35. In March 1998 approximately 77 aircraft were based at the airport. The reported number of operations per year is 22,400. All operations are by general aviation aircraft (Santos 1998).

The Tri-Cities Airport in Pasco is 34 mi to the southeast of the K Basins and CVDF and 29 mi from the CSB. The Federal Aviation Administration operates the air traffic control tower and airport radar approval control facility. The airport has two 7,700-ft crossing runways with 120°/300° and 030°/210° orientations. The latter has a 4,430-ft parallel runway. Runway 30 has a very high-frequency omnirange instrument approach, and runway 21R has an instrument landing system and is an instrument approach runway. In March 1998 about 108 aircraft were based at the airport. Total operations are reported as 74,000 per year with 68% general aviation, 29% commercial aviation (air taxis, air carriers), and 3% military aircraft. Commercial carriers operate a total of about 26 flights (arrivals and departures) per day.

Both NUREG-0800, Standard Review Plan, and DOE-STD-3014-96 provide methodology to evaluate and assess the significance of aircraft crash risk on facility safety. For the SNF Project, the methodology in DOE-STD-3014-96 is more conservative than that in NUREG-0800 because NUREG-0800 includes the assumption that protection provided in nuclear power plant
design for tornado-generated missiles is adequate for protection against general aviation light aircraft crashes. This is not a valid assumption for the CSB and CVDF. Therefore, the methodology of DOE-STD-3014-96 is used to evaluate the potential for aircraft crash into the CSB.

DOE-STD-3014-96 provides methodology to conservatively evaluate and assess the significance of aircraft crash risk on facility safety. The CVDF and CSB meet the standard's guidelines for applicability because they contain enough radioactive material to be classified as a hazard category 2 facilities according to the criteria established in DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480 23 Nuclear Safety Analysis Reports*.

The approach used in DOE-STD-3014-96 includes methodology to evaluate the frequency of aircraft impact into the facility. If the frequency of aircraft impact, calculated according to the methodology given in the standard exceeds $10^{-6}$/yr, the analysis proceeds to a consideration of whether those aircraft that have a high impact frequency could actually do damage resulting in potential releases from the facility. Using that methodology, the frequency of aircraft impact into the CVDF and CSB has been assessed (Beary 1997b).

The methodology provided in DOE-STD-3014-96 considers two contributors to the overall frequency of aircraft crashes: (1) the frequency of crashes resulting from nearby airport activity (i.e., takeoffs and landings) and (2) the frequency of crashes during overflights of the facility. Only runways within about 24 mi of the facility are included on the crash location probability tables given in Appendix B of DOE-STD-3014-96. Activities on runways at greater distances are not considered potential contributors to the frequency of a crash at the facility location for any category of aircraft. For calculating the frequency of crashes from aircraft flying over the site, DOE-STD-3014-96 provides DOE site-specific frequency data, in crashes per square mile per year centered at the site for various classes of aircraft.

For both the nearby airports and the overflight activities, DOE-STD-3014-96 bases the calculation of frequency of aircraft crash into a facility on a "four factor formula" that considers:

1. The number of aircraft operations at nearby airports and overflying the site (N)
2. The probability that an aircraft will crash (P)
3. The conditional probability that, given a crash, the aircraft crashes into a one-square-mile area where the facility is located ($f(x\ y)$)
4. The site-specific effective area of the facility ($A_{\text{eff}}$)

The formula is applied individually to each category of aircraft for both the nearby airports and the overflight activities. The overall frequency of aircraft crash into the facility is the sum of the frequencies calculated for all categories of aircraft and for all operational modes.
To evaluate the frequency of crashes that impact the facility from overflight operations, Tables B-14 and B-15 of DOE-STD-3014-96 give generic values of NFf(x,y) in crashes per square mile per year centered at various DOE sites, including the Hanford Site for five categories of aircraft. The values are based on the historic record for aircraft crashes in the continental United States.

The site-specific values of NFf(x,y) for overflight operations given for the Hanford Site by aircraft category are as follows:

- General aviation: $1 \times 10^{-4}$ crashes/yr/mi²
- Air carrier: $1 \times 10^{-7}$ crashes/yr/mi²
- Air taxi: $1 \times 10^{-6}$ crashes/yr/mi²
- Large military: $1 \times 10^{-7}$ crashes/yr/mi²
- Small military: $4 \times 10^{-8}$ crashes/yr/mi²

For both the nearby airport and the overflight operations, the NFf(x,y) for each category of aircraft is multiplied by an effective area, $A_{eff}$, representing the ground surface area within which an unobstructed aircraft, were it to crash within the area, would impact the facility either by flying or skidding into the facility. The effective area depends not only on the length, width, and height of the facility, but also on the aircraft's wingspan, the flight path angle, the heading angle relative to the heading of the facility, and the length of the skid. Therefore, an effective area is calculated for each category of aircraft.

Formulas for calculating the effective area of the facility are given in Appendix B of DOE-STD-3014-96. The effective area has two terms: the effective fly-in area $A_f$ and the effective skid area $A_s$.

$$A_{eff} = A_f + A_s$$

and

$$A_f = (WS + R) H \cot \phi + \frac{2 LW WS}{R} + LW$$

$$A_s = (WS + R) S$$

where

- $WS$ = aircraft wingspan
- $R$ = length of the diagonal of the facility $(L^2 + W^2)^{0.5}$
- $H$ = facility height
\[ \cot \phi = \text{mean of the cotangent of the aircraft impact angle} \]

\[ L = \text{facility length} \]

\[ W = \text{facility width} \]

\[ S = \text{aircraft skid distance} \]

See Sections 1-6 of the facility FSAR Annexes for the results of the calculations for the SNF Project facilities.

An analysis of the risk from helicopter accidents to the CSB and the CVDF was performed. The analysis was based on results of studies documented in the safety analysis reports for other Hanford Site facilities, Savannah River Site facilities, and for the NRC-licensed WNP-2.

Table 1-26 summarizes the studies examined, the study results, and the inferences the results provide relative to the SNF Project's design adequacy for protection against helicopter crashes.

The studies examined showed a consistently low frequency of impact and of penetration probability following impact (the latter despite the use of different algorithms for the calculation). Some of the studies addressed only crash frequencies and some addressed only damage probabilities. Three of the studies addressed aircraft in general but not helicopters specifically.

The combined results of the seven studies examined indicate a very conservative estimate of the frequency of helicopter crashes at close to \( 1 \times 10^{-6} \) per year. Those studies that specifically addressed penetration of the protective barriers showed this capability to be completely negligible. Therefore, the risk to SNF Project facilities of safety from helicopter crashes is very low. It can be conservatively concluded that the helicopter crash frequency is less than \( 1 \times 10^{-7} \) per year.

### 1-6.2 Other Transportation Accidents

A rail spur that passes by the K Basins and CVDF. This spur has not seen recent use and will not be used in support of K Basins Path Forward activities. The switch that would route trains on the spur is normally locked in the position that would route trains away from the 100 K Area (WHC-SD-WM-SAR-062).

Accidents that might occur on State Highway 240, such as explosions or toxic chemical releases, are judged to present a negligible risk to the SNF Project facilities because of the distance between the facility and the highway. At its closest approach, the distance is about 6 mi from the K Basins and CVDF and 5 mi from the CSB. NRC Regulatory Guide 1.78, *Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release* provides useful guidance on evaluating chemicals stored or situated at distances greater than 5 mi from the facility. It states that they need not be considered...
<table>
<thead>
<tr>
<th>Study</th>
<th>Relevance to SNF Project</th>
<th>Results</th>
<th>Conclusions relative to SNF Project</th>
</tr>
</thead>
<tbody>
<tr>
<td>Aircraft impact on CSB and CVDF</td>
<td>Very high</td>
<td>Crash frequency $1.43 \times 10^{-6}$/year</td>
<td>Product of frequency and penetration probability is very low</td>
</tr>
<tr>
<td>200 Area impact frequency</td>
<td>Modest</td>
<td>Crash frequency $&lt;1 \times 10^{-6}$</td>
<td>SNF Project crash frequency similar</td>
</tr>
<tr>
<td>Hanford Patrol aviation safety (helicopter)</td>
<td>Very high</td>
<td>Crash frequency and damage rate both very low</td>
<td>Gives high confidence of low crash and damage frequency at SNF Project facilities</td>
</tr>
<tr>
<td>WNP 2 aircraft crash</td>
<td>Modest</td>
<td>Low frequency of damaging crashes</td>
<td>Expect same low frequency at SNF Project facilities</td>
</tr>
<tr>
<td>HWVP helicopter safety</td>
<td>High</td>
<td>Penetration probability negligible</td>
<td>Penetration probability even less for SNF Project facilities</td>
</tr>
<tr>
<td>SRS helicopter crash</td>
<td>High</td>
<td>Combined frequency of strike and penetration very low</td>
<td>Implies that for SNF Project also low ($1 \times 10^{-7}$)</td>
</tr>
<tr>
<td>Hazardous facilities aircraft crash</td>
<td>High</td>
<td>Shows crash frequency for high overflight rate</td>
<td>Implies overall adequacy for SNF Project facilities</td>
</tr>
</tbody>
</table>

*Beary M M 1997 *Assessment of Aircraft Impact Frequency for the 200 Area Interim Storage Area* Contract No 000036 Task 14 Science Applications International Corporation Richland Washington
*WSRC TR 90 90 1990 *Analysis of the impact of SRS Helicopter Operations on Reactor Safety* Westinghouse Savannah River Company Aiken South Carolina
*See example helicopter calculation in DOE STD 3014 96 1996 *Accident Analysis for Aircraft Crash into Hazardous Facilities* U.S. Department of Energy Washington D C

CSB = Canister Storage Building
CVDF = Cold Vacuum Drying Facility
HWVP = Hanford Waste Vitrification Plant
SNF = spent nuclear fuel
SRS = Savannah River Site
WNP 2 = Washington Nuclear Power Plant No 2
because if a release occurs at such a distance, atmospheric dispersion will dilute and disperse the incoming plume to such a degree that there should be sufficient time for the operators to take appropriate action. In addition the probability of the plume remaining within a given sector for a long time is quite small. Because CSB is 5 mi from Highway 240 an evaluation was performed (see Section A 1 6 2 in Annex A).

NRC Regulatory Guide 1 78 also provides useful guidance for the evaluation of potential accidents involving hazardous chemicals that might be shipped past the SNF Project facilities on Route 4 North and the Hanford Site railroad. NRC Regulatory Guide 1 78 does not require control room habitability analysis for shipments less frequent than 10/yr for truck traffic, 30/yr for rail traffic or 50/yr for barge traffic. Neither the truck nor rail guidelines are exceeded for shipments of a quantity that could present a risk to the SNF Project facilities. Sections 3 3 of the facility FSAR Annexes address this hazard from the perspective of safety of workers and capability to maintain the facility in a safe state. Barge shipment on the Columbia River does not routinely occur above the Port of Benton barge facility as discussed in Section 1 3 1.

Regulatory Guide 1 91, Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants, describes a method for determining distances from critical plant structures beyond which any explosion that might occur on a railway, highway, or navigable waterway is not likely to have an adverse effect on plant operation or to prevent safe shutdown. The method is based upon an NRC staff judgement that for structures of concern an acceptable overpressurization limit from such explosions can be conservatively chosen at 1 lb/in². Although not stated in the Regulatory Guide, it is assumed that the 1 lb/in² value was established for structures that are designed to withstand the severe natural phenomena loadings that are typical for safety-related structures of nuclear power plants.

1 7 NEARBY FACILITIES

See Sections 1 7 of the facility FSAR Annexes for discussions of facilities near the SNF Project facilities.

1 8 VALIDITY OF EXISTING ENVIRONMENTAL ANALYSES

No significant discrepancies have been identified between the site characteristic assumptions made in this chapter and those made in the SNF Project Environmental Impact Statement (DOE/EIS-0245F).

1 9 REFERENCES

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DOE Order 6430 1A *General Design Criteria* U S Department of Energy, Washington D C


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PNL-4622, 1983 *Climatological Summary for the Hanford Area*, Pacific Northwest Laboratory, Richland, Washington


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WHC-SD-EN-TI-011, 1992 *Geology of the Northern Part of the Hanford Site An Outline of Data Sources and the Geologic Setting of the 100 Areas* Rev 0, Westinghouse Hanford Company, Richland, Washington


WHC-SD-NR-ER-093, 1992 *Foundation Studies WNP-1 100 N Site* Rev 0 Westinghouse Hanford Company, Richland Washington


WHC-SD-SNF-DB-009, 1996 *Canister Storage Building Natural Phenomena Hazards* Rev 4 Westinghouse Hanford Company Richland Washington


Figure 1-1  Location of the Hanford Site in the State of Washington

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Figure 1-2  100 K Area Location Map
Figure 1-3  Hanford Site Boundaries

- Wehiuke Wildlife Recreation Area, State of Washington Department of Game
- Saddle Mountain National Wildlife Refuge, U.S. Fish and Wildlife Service
- Fitzner-Eberhardt Arid Lands Ecology Reserve
- State of Washington Leased Land

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Figure 1-4 The Hanford Site, Counties, and the Regional Highway Network
Figure 1-5  Hanford Site and Surrounding Area
Figure 1-6 Population of Cities and Counties Near the Hanford Site

Note: If population is not listed, it is less than 1,000.

HMS = Hanford Meteorological Station
Figure 1-7 Distribution of Transient Population

Key

midway between WNP 1, 2, and 4

migratory agricultural workers

recreational

To convert miles to kilometers, multiply by 1.6093
Figure 1-8 Hanford Meteorological Network

Station Number  Station Name                      Station Number  Station Name
2. Emergency Operations Center   15. Franklin County
3. Army Loop Road                16. Gable Mountain
4. Rattlesnake Springs           17. Ringold Formation
5. Edeh                        18. Richland Airport
6. 200 East Area                19P. Plutonium Finishing Plant
7. 200 West Area                19S. Segrel (inactive)
8B. Beverly                     20. Rattlesnake Mountain
9W. Washuk Slope (inactive)      21. Hanford Meteorological Station (125 m)
10. Yakima Barricade             23. Gable West
11. 300 Area (80 m)              24. 100 F
13. 100 N (60 m)                 26. Benton City
17. 200 East Area

Note: All network stations are 91.4 m unless otherwise indicated.

To convert meters to feet, multiply by 3.281
Figure 1-9  Hanford Site in Relation to Surrounding Terrain
Figure 1-10  Six-Hour Precipitation Hazard Curves for the Hanford Site

- Stone et al. 1983
- Hansen et al. 1994
- 1 mi² storm
- 10 mi² storm


Figure 1-11 Greatest Depth of Snow on the Ground at Hanford from 1946-47 to 1980-81

Greatest depth was less than 1 in during the winter of 1957. 1958
To convert inches to centimeters, multiply by 2.54
Figure 1-12 Wind Hazard Curves with Design Windspeeds
(Straight Wind Windspeeds are Fastest Mile)

Coats, D W and R C Murray 1985 Natural Phenomena Hazards Modeling Project Extreme Wind/Tornado Hazard Models for Department of Energy Sites, UCRL 53526 Rev 1 Lawrence Livermore National Laboratory Livermore California.

Figure 1-13 Location of Surface Water on the Hanford Site
Figure 1-14 Location of Dams on the Columbia River

Columbia River drainage basin
○ dams on Columbia River

Map Index Number
1 Bonneville 8 Rocky Reach
2 The Dalles 0 Wells
3 John Day 10 Chief Joseph
4 McNary 11 Grand Coulee
5 Priest Rapids 12 Hugh Keenleyside
6 Wanapum 13 Revelstoke
7 Rock Island 14 Mica

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Figure 1-15  Extent of Probable Maximum Flood in Cold Creek Area
Figure 1-16 Characteristic Stratigraphy, Lithology, and Hydrogeologic Conditions on the Hanford Site

- Hanford formation
- Ringold Formation
- Elephant Mountain member
- Rattlesnake Ridge Interbed
- Early Paleocene / Fife-Paradise unit

Legend:
- gravel
- sand
- silt
- cemented calcium carbonate (caliche)
- water table
- locally confined aquifer
- basalt and interbed aquifer system
- vadose zone

Thickness:
- 0 to 80 m
- 0 to 15 m
- 0 to 21 m
- 0 to 35 m
- 37 to 104 m

To convert meters to feet, multiply by 3.281
Figure 1-17  Water Table Contours for the 100 K Area, June and September 1994
Figure 1-18 Location of the Hanford Site, Pasco Basin and Columbia Plateau
Figure 1-19 Physiographic Provinces of the Pacific Northwest with the Columbia Plateau in White
Figure 1-20 Generalized Stratigraphy of the Pasco Basin

<table>
<thead>
<tr>
<th>Period</th>
<th>Epoch</th>
<th>Group</th>
<th>Formation</th>
<th>Member (formal and informal)</th>
<th>Sediment stratigraphy or basalt flows</th>
</tr>
</thead>
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<tr>
<td>Quaternary</td>
<td>Holocene</td>
<td>Plio-Pleistocene Interval</td>
<td>Hanford Formation</td>
<td>member of Savage Island</td>
<td>Sand, silt, clay, and basalt flows</td>
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<td>member of Taylor Flat</td>
<td>Landslides and basalt flows</td>
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<td>member of Wooded Island</td>
<td>Tuff, ash, and basalt flows</td>
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<td>Ringold Formation</td>
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<td>Sedimentary Basalt</td>
<td>8.5 Ice Harbor Member</td>
<td>basalt of Goose Island</td>
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<td>basalt of Martindale</td>
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<td>basalt of Basin City</td>
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<td>basalt of Ward Gap</td>
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<td>basalt of Elephant Mountain</td>
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<td>Rattlesnake Ridge interbed</td>
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<td>basalt of Pomona</td>
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<td>Selah interbed</td>
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<td>basalt of Gable Mountain</td>
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<td>Cold Creek interbed</td>
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<td>basalt of Running</td>
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<td>Quincy interbed</td>
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<td>Squaw Creek interbed</td>
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<td>basalt of Lyon Ferry</td>
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<td>basalt of Sentinel Gap</td>
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<td>basalt of Sand Hollow</td>
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<td>basalt of Silver Falls</td>
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<td>Vantage interbed</td>
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<td>basalt of Rocky Coulee</td>
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<td>basalt of Birkett</td>
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<td>member of Sentinal Bluffs</td>
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<td>member of Umatum</td>
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<td>member of McCoy Canyon</td>
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<td>basalt of Umatum</td>
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<td>member of Steep Canyon</td>
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<td>member of Otley</td>
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<td>member of Grouse Creek</td>
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<td>member of Wapahilla Ridge</td>
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<td>member of Mt. Horrible</td>
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<td>member of China Creek</td>
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<td>member of Teepse Bluffs</td>
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<td></td>
<td></td>
<td>member of Buckhorn Springs</td>
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<td>member of Rock Creek</td>
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<td></td>
<td></td>
<td>member of American Bar</td>
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</tr>
</tbody>
</table>

*The Grande Ronde Basalt consists of at least 120 major basalt flows comprising 17 members

N2, N3, N4, and N5 are magnetostatigraphic units.
Figure 1-21  Idealized Suprabasalt Subsurface Stratigraphy of the Hanford Site

Explanation of symbols and abbreviations used in cross section

<table>
<thead>
<tr>
<th>Grain size scale, indicative dominant grain size</th>
</tr>
</thead>
<tbody>
<tr>
<td>- boulder-gravel</td>
</tr>
<tr>
<td>- gravel-cobble gravel</td>
</tr>
<tr>
<td>- clay-to-coarse-grained sand</td>
</tr>
<tr>
<td>- clay and silt</td>
</tr>
</tbody>
</table>

Subordinate lithologies and other lithologic symbols

<table>
<thead>
<tr>
<th>D</th>
<th>gravel-rich</th>
</tr>
</thead>
<tbody>
<tr>
<td>C</td>
<td>illitic-rich</td>
</tr>
<tr>
<td>B</td>
<td>basalt</td>
</tr>
<tr>
<td>A</td>
<td>sand-rich</td>
</tr>
<tr>
<td></td>
<td>pedogenic carbonate</td>
</tr>
</tbody>
</table>

Stratigraphic Units

- Hanford Formation
- Plio-Pleistocene Palouse soil
- Pre-Missoula gravels
- Ringold upper sands and silts
- Ringold sub E unit
- Ringold unit C
- Ringold sub C unit
- Ringold unit B/D
- Ringold Lower Mud unit
- Ringold unit A
- sub Ringold fine sediments
- basalt

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Figure 1-22  Geologic Cross Sections of the 100 K Area
(sheet 1 of 3)

100-K Area

K-32B  Borehole Number
(Those numbers beginning with "K"
have the prefix "199",
i.e., 199 K 22)

- Borehole Location
  (existing well)

○ Borehole Location
  (new well)

--- Location of Geologic
  Cross-Section

0  200  400 Meters

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Figure 1-22 Geologic Cross Sections of the 100 K Area
(sheet 2 of 3)

Legend
- Formation Contacts
- Unit or Sequence Contact (Dashed where inferred)
- Water Table
- Secondary Features
  - Gravels
  - Sand
  - Silty or Clayey
  - Fill
- Primary Features
  - Silts or Clays
  - Sand
  - Boulders

Horizontal Scale
- 10 K (100 feet)
- 15-20 K (90 feet)
- Vertical Scale
- 10 K (100 feet)

Elevation
- North
- South
Figure 1-22 Geologic Cross Sections of the 100 K Area (sheet 3 of 3)
Figure 1-23  Annual Probability of Volcanic Ash Accumulation on the Hanford Site

Greater than 1 0 cm (0.4 in)

Greater than 10 0 cm (4.0 in)
Figure 1-24 Structural Map of the Yakima Fold Belt
Figure 1-25 Generalized Geologic Map of the Pasco Basin

Generalized geologic map of the Pasco Basin

- Late Cenozoic sediments (surficial Quaternary sediments)
- Hanford Formation
- Ringold Formation
- Saddle Mountains Basalt
- Wanapum Basalt
- Grande Ronde Basalt

Legend:
- Fault
- Anticline
- Syncline

Elevation
- Feet
- Meters

MSL = Mean Sea Level

[Map showing geological formations and topography of the Pasco Basin]
Figure 1-26  Generalized Location of Earthquake Swarms in the Pasco Basin

BEN = Benson Ranch
BRV = Black Rock Valley
BVW = Beverly
CRF = Corfu
ET3 = Eltopla Three
GBB = Gable Butte
GBL = Gable Mountain
H2O = Water
LOC = Locke Island
MDW = Midway
MJ2 = May Junction Two
OT3 = Othello Three
PRO = Prosser
RC1 = Royal City One
RED = Red Mountain
RSW = Rattlesnake Mountain
SNI = Snively Ranch
WA2 = Wahluke Slope
WG4 = Wallula Gap Four
WIW = Wooded Island
WRD = Warden

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Figure 1-27  Earthquakes at the Hanford Site 1990 through 1995
Figure 1-28  Magnitude 3 or Greater Earthquakes Recorded at the Hanford Site from 1969 through 1995
Figure 1-29  Seismic Hazard Curves for the Cold Vacuum Drying Facility Location

![Seismic Hazard Curves](image-url)
Figure 1-30 Contribution of the Various Folds to the Mean Hazard from the Yakima Fold
Figure 1-31 Contributions of the Crustal and Cascadia Subduction Zone Sources to the Mean Seismic Hazard at the 100 K Area
Figure 1-32 Contributions of the Three Crustal Sources to the Mean Seismic Hazard from All Crustal Sources at the 100 K Area
Figure 1-33  Cold Vacuum Drying Facility Performance Category 3
Response Spectra

5% Damping

- Horizontal Response Spectrum
- Vertical Response Spectrum

Spectral Acceleration Amplification (g)

Hz

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Figure 1-34 Cascade Range Volcanic Ash Hazard

To convert centimeters to inches multiply by 0.3937


7998030008 44 CVDF

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Figure 1-35  Federal Airways and Instrument Approaches and Departures

Analysis: The diagram illustrates the Federal Airways and Instrument Approaches and Departures, providing a visual representation of various navigation checkpoints and routes. The map includes symbols for airports, cities, and other landmarks, with directional indicators and compass headings denoted.
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CHAPTER 20

FACILITY DESCRIPTION
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   2.1 INTRODUCTION
   2.2 REQUIREMENTS
   2.3 FACILITY OVERVIEW
      2.3.1 K Basins Spent Nuclear Fuel
      2.3.2 Other Spent Nuclear Fuel
   2.4 FACILITY STRUCTURE
   2.5 PROCESS DESCRIPTION
      2.5.1 Technical and Safety Discussion for K Basins Spent Nuclear Fuel
      2.5.2 Spent Nuclear Fuel Project Authorization Basis
   2.6 CONFINEMENT SYSTEMS
   2.7 SAFETY SUPPORT SYSTEMS
   2.8 UTILITY DISTRIBUTION SYSTEMS
   2.9 AUXILIARY SYSTEMS AND SUPPORT FACILITIES
   2.10 REFERENCES

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2-2 Spent Nuclear Fuel Subproject Safety Interfaces
2-3 Multi-Canister Overpack Pressurization (Bounding Safety Case)
2-4 Spent Nuclear Fuel Project Facilities and Process Physical Barriers
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2-2 Applicability of Potential Hazards to Spent Nuclear Fuel Project Processes, and Technical Analyses Required to Evaluate Effects of Potential Hazards 2-8

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LIST OF TERMS

BPA   Bonneville Power Administration
CSB   Canister Storage Building
CVDF  Cold Vacuum Drying Facility
DOE   U S Department of Energy
FSAR  final safety analysis report
ISA   Interim Storage Area
MCO   multi-canister overpack
MTU   metric ton of uranium
SNF   spent nuclear fuel
SPR   single pass reactor
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20 FACILITY DESCRIPTION

21 INTRODUCTION

This chapter provides an integrated overview description of the final hazard category 2 Spent Nuclear Fuel (SNF) Project facilities, processes, and operations. Facility-specific descriptions of the individual SNF Project facilities and processes are provided in the facility annexes (Annex A, Canister Storage Building [CSB], Annex B, Cold Vacuum Drying Facility [CVDF] and Annex D, 200 East Area Interim Storage Area [ISA]) to this SNF Project Final Safety Analysis Report (FSAR). Descriptions of the SNF Project-related facilities and processes at the K Basins are provided in WHC-SD-WM-SAR-062 K Basins Final Safety Analysis Report.

22 REQUIREMENTS

The requirements that form the basis for the design of SNF Project facilities are found in HNF-SD-SNF-RD-001, Spent Nuclear Fuel Project Standards/Requirements Identification Document. Specific requirements applicable to this chapter include the following:

- U.S. Department of Energy (DOE) Order 5400.1, General Environmental Protection Program
- DOE Order 5480.7A, ‘Fire Protection’
- DOE Order 5480.24, Nuclear Criticality Safety
- DOE Order 5480.28, Natural Phenomena Hazards Mitigation
- DOE Order 6430.1A, General Design Criteria
- ANSI/ACI 349-85, Code Requirements for Nuclear Safety Related Concrete Structures

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- ANSI/AISC N690-94 *Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities*
- ANSI/ANS-57 1-1992 *Design Requirements for Light Water Reactor Fuel Handling System*
- ANSI/ANS-57 9-1992, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)*
- ANSI/ASME 30 17 *Overhead and Gantry Cranes (Top Running Bridge, Single Girder, Underhung Hoist)*
- ANSI/ASME B30 10, *Safety Standard for Hooks*
- ANSI/ASME B30 16, *Overhead Hoist (Underhung)*
- ANSI/ASME HST-1M, *Performance Standard for Overhead Electric Wire Chain Hoists*
- ASCE-7-93 *Minimum Design Loads for Building and Other Structures*
- ASME B30 20-1993 *Below-the-Hook Lifting Devices*
- ASME *Boiler and Pressure Vessel Code (ASME 1995)*
- ASTM C 1055 *Standard Guide for Heated System Surface Conditions that Produce Contact Burn Injuries*

Additional requirements that apply only to a specific facility are identified in the applicable facility FSAR Annex.
A requirement for new SNF Project facilities to achieve nuclear safety equivalency comparable to US Nuclear Regulatory Commission licensed facilities was established in 1995 (Sellers 1995) This requirement has been implemented for the CVDF and CSB Further discussion of these requirements is provided in Section 3.3 of this SNF Project FSAR

2.3 FACILITY OVERVIEW

The SNF Project is involved with consolidation and safe interim storage of the spent fuels that are located on the Hanford Site These fuels are treated as two groups The first is that group know as K Basins SNF This group requires retrieval from underwater storage, cleaning, packaging into storage containers, dewatering, inerting, and transport to the interim storage location. The second group is the SNF currently dry-stored in several locations on the Hanford Site This group will be transported to a common dry storage location in the 200 East Area

For the K Basins SNF WHC-SD-SNF-SP-005 Hanford Spent Nuclear Fuel Project Integrated Process Strategy established the technical framework for constructing facilities and implementing processes that are compatible with DOE goals. This process strategy has subsequently evolved to the current technical baseline based on implementation of the Baseline Change Request, SNF-98-006, Simplified SNF Project Baseline SNF-98-006 describes process and design approaches for moving SNF from the K Basins in water-filled multi-canister overpack (MCO) containers, vacuum drying the fuel at the CVDF, a new facility constructed near the K Basins, and moving the SNF in sealed MCOs to the CSB, a new dry interim storage facility located in the Hanford Site 200 East Area

2.3.1 K Basins Spent Nuclear Fuel

New systems, equipment, and facilities are being constructed and modifications are being made to the K Basins to support the removal of the K Basin spent fuel sludge, and debris. The new systems include the fuel retrieval system, sludge removal system, debris removal system, integrated water treatment system, and cask loadout system. Modification upgrades are also being made to the K Basins to allow installation of these systems and to provide related infrastructure support. The integrated water treatment system upgrade will handle the increased need for treated water to support the new systems and to maintain basin water quality during in-basin activities. The end result of these processes is SNF loaded into water-filled MCOs and moved out of the K Basins. Detailed descriptions of the K Basin systems and processes are provided in the K Basin FSAR (WHC-SD-WM-SAR-062)

The CVDF has been constructed near the K Basins to accept loaded water-filled MCOs and to remove the free water. The CVDF contains process bays in which water is removed from the MCOs. CVDF processing systems control the temperature of the MCO contents, minimize oxygen concentrations via helium inerting, establish vacuum and purge for drying, and use controlled and filtered ventilation to control emissions. Detailed descriptions of the CVDF systems and processes are provided in Annex B to this SNF Project FSAR
The CSB has been constructed in the 200 East Area to accept dry MCOs for interim storage of up to 40 years. The CSB will store the MCOs in vertical tubes in the first of three below-ground vaults. The CSB contains processing systems for receipt and handling of the MCOs, sampling of a selected number of MCOs, welding permanent caps on the MCOs, and heating, ventilation and air conditioning to control temperature and monitor emissions. Detailed descriptions of the CSB systems and processes are provided in Annex A to this SNF Project FSAR.

The MCO is a key component in the packaging, transport, drying, and interim storage processes for the K Basin SNF. The MCO is a cylinder, approximately 14 ft long and 2 ft in diameter. Each MCO is capable of holding up to six baskets of fuel and scrap. Approximately 400 MCOs will be used for interim storage of the SNF. The MCO is of robust design, capable of withstanding various drop scenarios and containing internal pressures of up to 450 lb/in² gauge with the welded cover cap in place. As a prudent action, monitoring a small number of the first MCOs shipped to the CSB for a period of up to 2 years will provide data indicating the internal gas generation rate and pressure buildup to show that the SNF Project process meets long-term storage requirements.

The MCO provides primary confinement for the SNF during processing and storage. In the CVDF it serves as part of the primary processing vessel. Because the MCO plays an integral part in the design, operations, and safety functions for the entire SNF Project, a separate document has been prepared to describe the MCO design and technical issues in an integrated fashion for the entire SNF Project. This detailed description of the MCO is provided in HNF-SD-SNF-SARR-005 Multi-Canister Overpack Topical Report.

Transportation of the MCOs from the K Basins to the CVDF and from the CVDF to the CSB is described in HNF-SD-TP-SARP-017, Safety Analysis Report for Packaging (Onsite) Multi-Canister Overpack Cask. The packaging and venting requirements during transport are addressed in the facility annexes to this SNF Project FSAR. The interface with transport functions at each facility and the transport of the MCO between facilities is addressed in the SNF Project FSAR with annexes or the K Basins FSAR (WHC-SD-WM-SAR-062).

2.3.2 Other Spent Nuclear Fuel

The 200 East Area ISA has been constructed a few hundred feet west of the CSB in the 200 East Area. The ISA is a relatively simple facility consisting of concrete and gravel pads with fencing and lighting. The ISA will be used to store non-defense reactor SNF housed in above-ground dry cask storage systems. Three different SNF types will be stored in three different dry cask systems. Detailed descriptions of the ISA, SNF types, and dry cask systems are provided in Annex D to this SNF Project FSAR.
2 4 FACILITY STRUCTURE

For an overview of the basic facility-specific buildings and structures, including construction details such as basic floor plans, equipment layout, construction materials, controlling dimensions, and dimensions significant to the hazard and accident analyses, see the facility FSAR Annexes or the K Basins FSAR (WHC-SD-WM-SAR-062).

2 5 PROCESS DESCRIPTION

The major process steps involved in moving K Basin SNF to interim storage are shown in Figure 2-1. At the K Basins, spent fuel assemblies are removed from the existing canisters, cleaned and placed into specially designed baskets that are loaded into an MCO. The MCO is filled with water, the MCO shield plug is installed, and the MCO is transported in a cask to the CVDF. At the CVDF, the MCO is drained of water, vacuum dried, leak tested and inerted with helium. The mechanical sealing of the MCO is completed and the MCO is leak tested and transported to the CSB. At the CSB, the MCO is received, and those that have been designated for sampling over a 2-year time span are moved to storage tubes. All other MCOs are queued for permanent sealing, which is achieved by welding an end cap in place. An MCO designated for sampling will have an end cap welded in place at the end of the sampling period. After welding, the MCO is placed in interim storage within long vertical tubes.

Non-defense reactor SNF currently dry-stored in several locations on the Hanford Site will be transported to the ISA and stored in above-ground dry cask storage systems.

2 5 1 Technical and Safety Discussion for K Basins Spent Nuclear Fuel

2 5 1 1 Safety Interfaces Between Facilities. Figure 2-2 shows the key SNF subproject safety interfaces and identifies the major safety assumptions applicable to those interfaces. The figure shows the safety analysis documents (HNF-3553 Annex A, Annex B and Annex D and WHC-SD-WM-SAR-062) outlined with a dotted line where the major interface issues are discussed. The MCO Topical Report HNF-SD-SNF-SARR-005) also is shown as technical support to the K Basin FSAR and to both the CSB FSAR (Annex A) and the CVDF FSAR (Annex B). While the safety analysis report for packaging (HNF-SD-TP-SARP-017) is not explicitly shown in the figure, the major safety assumptions of the report are included with the major process flow arrows between facilities. The K Basins processes and fuel transportation controls ensure that the MCOs transferred from the K Basins to the CVDF meet the following interface criteria:

- Limit the number of scrap baskets to two per MCO
- Load the MCO in accordance with the loading procedures
- Follow the prescribed cleaning procedures
- Keep transit time within the prescribed 24-hour shipping window during shipment of the MCO to the CVDF

- Follow the prescribed MCO drying, testing, and cask inerting and sealing procedures

During the drying process at the CVDF, the mass of free water is reduced to less than 0.2 kg. This mass is not actually measured at the completion of drying but process controls and the results of proof-of-dryness testing provide evidence that the drying process has been successful. A mathematical correlation was developed and calibrated using test data that modeled residual free water after vacuum drying (HNF-1851). This correlation and the process controls for temperature and pressure, which ensure conditions consistent with the model and no introduction of additional water, are adequate to meet the interface criteria of <0.2 kg of free water. A proof-of-dryness test is used to demonstrate that initial conditions for the correlation are met and fuel reaction rates are limited.

The safety analysis for the CSB is based on upper limits of particulate and water in the MCO. Evaluations were made for low nominal and upper limits of the residual particulate mass, generated particulate mass, and bound water (HNF-1527, HNF-1523). The worst-case conditions have been used in the safety analyses. Accidents at the CSB are based on 64 kg of particulate and 464 kg of bound water (in addition to free water as discussed below). The upper limit on residual particulate assumes no effectiveness for the cleaning process.

2.5.1.2 Key Spent Nuclear Fuel Project Safety Considerations. The SNF Project safety basis and the related safety analysis report accident analyses for the K Basins, CVDF, and CSB consider the following:

- MCO and confinement barrier integrity (including drops)
- MCO overpressurization
- Fuel reactivity and corrosion behavior
- Hydrogen generation and potential for deflagration or detonation
- Oxygen generation and reactivity with MCO contents
- Product end state and ability to seal the MCO (applies for interim storage)

Technical issues have been identified through a combination of systems engineering safety analysis and technology acquisition program activities. To support the SNF Project’s schedule objectives, the SNF Project is organized into several subprojects. However, the major technical issues tend to cut across many of the subprojects. Each of the SNF Project facility FSAR Annexes and the K Basins FSAR (WHC-SD-WM-SAR-062) address a comprehensive set of equipment failures, design and construction defects, operator errors, natural phenomena hazards (e.g., seismic, wind), external events (e.g., fire), and process hazards that could challenge safety barriers or defense-in-depth mitigating features.
Table 2-1 identifies the process characteristics and related potential hazards that drive the safety analysis for the SNF Project involving K Basin spent fuel handling, processing, and storage. Six major categories of process-related hazards are identified in Table 2-1.

Table 2-1 Process Characteristics and Potential Hazards that Drive Spent Nuclear Fuel Project Safety Analyses

<table>
<thead>
<tr>
<th>Process characteristics</th>
<th>Potential hazard</th>
<th>Accident condition</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fissile content of fuel and/or sludge</td>
<td>Criticality ($k_{eff} \geq 0.95^a$ or 0.98$^b$)</td>
<td>• Unacceptably high $k_{eff}$</td>
</tr>
<tr>
<td>Chemical reactivity of the fuel</td>
<td>•Rapid $U-H_2O$ reaction$^c$</td>
<td>MCO overpressurization</td>
</tr>
<tr>
<td></td>
<td>•Rapid $U-O_2$ reaction$^c$</td>
<td>• MCO thermal runaway reaction</td>
</tr>
<tr>
<td></td>
<td>Slow $U-H_2O$ reaction</td>
<td>$H_2$ deflagration resulting in MCO or tube overpressurization</td>
</tr>
<tr>
<td>Radiolysis</td>
<td>•$H_2-O_2$ reaction$^d$</td>
<td>MCO overpressurization</td>
</tr>
<tr>
<td></td>
<td>•$H_2O$ decomposition</td>
<td>$H_2$ deflagration resulting in MCO or tube overpressurization</td>
</tr>
</tbody>
</table>

$k_{eff}$ for new facilities

$^a$k eff for existing facilities (at K Basins until spent fuel is loaded into MCO)

Rapid reactions can also include $UH_4$ reactions with $H_2O$ or $O_2$.

$^d$ $H_2$ and $O_2$ can react explosively with each other and can react with MCO contents to form hydrides and oxides.

MCO = multi canister overpack

The retained water inventory (free water plus chemically bound water) drives the process control strategy for all of the accidents except nuclear criticality. Process and administrative controls are established to optimize process efficiency and to maintain process operations within an acceptable safety envelope. Analyses have been performed and documented that address each of the major process-related hazards. Models have been developed and conservative enabling assumptions have been established when necessary to allow analysis to proceed without impacting project schedules. Characterization and testing required to verify enabling assumptions have been performed and documented.

Nuclear criticality controls are in place to maintain $k_{eff}$ less than or equal to 0.98 in the K Basins or 0.95 in the MCO and the new SNF Project facilities. These controls include mass and geometry controls in the K Basins and CVDF processing and geometry and moderator control in the CSB.

Table 2-2 relates the potential hazards identified in Table 2-1 with the steps used in processing the K Basin SNF and identifies the technical analyses that are needed to address these hazards and to estimate the effect of these hazards on the process. The effects on the processes include criticality radiolysis, hydrogen deflagration and detonation reactions with the fuel.
and sludge drying curves MCO pressurization, thermal-hydraulic analysis, and reactant consumption and gettering

Table 2-2  Applicability of Potential Hazards to Spent Nuclear Fuel Project Processes, and Technical Analyses Required to Evaluate Effects of Potential Hazards

<table>
<thead>
<tr>
<th>Potential hazard</th>
<th>Fuel retrieval</th>
<th>Fuel transfer</th>
<th>CVDF</th>
<th>Ship to CSB</th>
<th>Store in CSB</th>
<th>Technical analyses needed to address the hazards</th>
</tr>
</thead>
<tbody>
<tr>
<td>Criticality ($k_{ef} \geq 0.95$ or $0.98^b$)</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>Criticality analysis</td>
</tr>
<tr>
<td>Rapid U–H$_2$O reaction</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>U H$_2$O reaction analysis</td>
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<td></td>
<td></td>
<td></td>
<td>MCO pressurization analysis</td>
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<td></td>
<td></td>
<td></td>
<td></td>
<td>Thermal hydraulic analysis</td>
</tr>
<tr>
<td>Rapid U–O$_2$ reaction</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>U O$_2$ reaction analysis</td>
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<tr>
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<td></td>
<td>Thermal hydraulic analysis</td>
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<td></td>
<td></td>
<td>MCO pressurization analysis</td>
</tr>
<tr>
<td>Slow U–H$_2$O reaction</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
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<td>U H$_2$O reaction analysis</td>
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<td>Thermal hydraulic analysis</td>
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<td></td>
<td>MCO pressurization analysis</td>
</tr>
<tr>
<td>$H_2$–O$_2$ reaction$^d$</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>$H_2$ deflagration reaction analysis</td>
</tr>
<tr>
<td></td>
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<td></td>
<td></td>
<td></td>
<td></td>
<td>Uranium hydride formation analysis</td>
</tr>
<tr>
<td>$H_2$O decomposition</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td></td>
<td>Radiolysis analysis</td>
</tr>
</tbody>
</table>

$^a$k$_{ef}$ for new facilities

$^b$k$_{ef}$ for existing facilities (at K Basins until spent fuel is loaded into MCO)

Rapid reactions can also include UH$_x$ reactions with H$_2$O or O$_2$.

$^d$H$_2$ and O$_2$ can react explosively with each other and can react with MCO contents to form hydrides and oxides.

CSB = Canister Storage Building
CVDF = Cold Vacuum Drying Facility
MCO = multi-canister overpack

2.5.1.2.1 Multi-Canister Overpack Integrity  Maintaining MCO integrity is a primary safety function. The two principal risks to MCO integrity are MCO overpressurization and deflagration or detonation within the MCO. Only these two key technical issues, which have required characterization or testing to sufficiently resolve important technical uncertainties, are summarized here. Other issues and assumptions relating to the safety basis are discussed in the facility FSAR Annexes.

Multi-Canister Overpack Pressure Retention  The pressure retention function is one of the primary safety features of the MCO. Pressure buildup can be due to temperature rise,
radiolytic gas generation, or fuel corrosion reactions. However, the principal source of pressure rise is hydrogen gas generated during a water-uranium corrosion reaction. Water inventory in the MCO can come from free water, waters of hydration of several MCO and fuel element constituents, and chemically bound water. The principal focus of the SNF Project processes is to limit the particulate as a source of waters of hydration and chemically bound water and to remove most of the free water.

Figure 2-3 reflects the MCO pressurization safety basis summary. This figure depicts the amounts of particulate loaded in an MCO at K Basins and generated during transportation and drying, the amounts of water in an MCO after drying in the CVDF, and MCO pressure contributions from MCO contents stored for 40 years. The first column in Figure 2-3 shows the total particulate content (64 kg) within a bounding MCO and the way in which it is distributed between generated particulate, aluminum hydroxide and uranium oxide (HNF-1527). The center column of Figure 2-3 shows how the particulate in the first column contributes to the water content (4.84 kg) of an MCO shipped to the CSB (HNF-1523). Notice that the aluminum hydroxide (hydrated water) contributes the largest amount of water. Also included in the water total is the 200 g of residual water estimated as an upper bounding limit to remain in the MCO after CVDF drying operations. The last column in Figure 2-3 depicts the estimated pressure for an MCO after 40 years of storage (62 lb/in² gauge), showing the relative amounts of pressure contributed by the internal water hydrates, and aluminum hydroxide components (HNF-SD-SNF-TI-040). Estimates from the nominal calculations show an expected pressure of less than 16 lb/in² gauge. The third column also shows the MCO design margins present for the mechanically sealed MCO (150 lb/in² gauge) and the welded MCO (450 lb/in² gauge).

Hydrogen and Oxygen Concentrations in a Multi-Canister Overpack. Following cold vacuum drying, the MCO is sealed for interim storage with a helium atmosphere slightly above atmospheric pressure. Hydrogen is generated by radiolytic and chemical processes. Hydrogen concentrations over the 40-year storage period are expected to be significant. Oxygen is also generated by radiolysis of aluminum hydroxide and metal oxide hydrates (U, Fe, Al) that are contained in an MCO.

Lower flammability limits of oxygen and hydrogen were established as a safety basis criterion for acceptable oxygen buildup. Since oxygen is consumed by reaction with exposed fuel elements and fuel scrap, the safety basis strategy has been to demonstrate that inherent oxygen gettering within the MCO sufficiently reduces the oxygen concentration and therefore the risk of deflagration or detonation. Exposed uranium is the principal oxygen gettering material within the MCOs. Consequently, the number of scrap baskets within an MCO is a key variable related to the oxygen gettering scenarios. The worst-case scenario is an MCO containing intact fuel with no exposed uranium and zero scrap baskets. As stated in HNF-SD-SNF-TI-040, MCO Internal Gas Composition and Pressure During Interim Storage, there is no combination of high MCO power, water content, and reaction area that results in oxygen concentrations above the lower flammability limit. Under best-estimate water inventories, a worst-case oxygen concentration of no more than 0.75% is predicted. It was concluded that oxygen concentrations throughout the 40-year storage period for MCOs containing either zero, one, or two scrap baskets will not reach.
the minimum concentration needed to support combustion (4%) The technical basis for the oxygen gettering safety basis is addressed in HNF-SD-SNF-TI-040

2 5 1 2 2 Summary of Accident Scenarios and Preventative and Mitigative Features

Key facilities and process physical barriers have been identified and evaluated for K Basins SNF retrieval MCO transport and processing and interim storage This barrier evaluation identified the accidents of concern that are applicable for each of the major process elements The key prevention mitigation or control features identified in Figure 2-4, are required to sufficiently reduce the risk of the important safety accidents for each major process element Monitoring and measurements also identified in Figure 2-4 provide the feedback needed to ensure that treatment transport or storage operations are maintained within the established operating design basis envelopes However the monitoring system is not a safety basis feature

The identified safety envelopes system design features and process and administrative controls establish sufficient safety margins to provide safe and environmentally sound management of K Basin SNF The facility FSARs for the CSB (HNF-3553 Annex A) CVDF (HNF-3553 Annex B) and K Basins (WHC-SD-WM-SAR-062) define the equipment and control features needed to prevent or mitigate the consequences of accidents

2 5 1 3 Technical Documentation

The fundamental technical data used to support the safety basis for the processing and sealed storage of MCOs is documented in HNF-SD-SNF-TI-015, Spent Nuclear Fuel Project Technical Databook Data required for support of the safety basis that is not available through literature review has been developed through an SNF Project engineering studies characterization efforts and process equipment test programs Although hundreds of documents have been produced the Technical Databook (HNF-SD-SNF-TI-015) directly references 23 documents that provide key technical baseline support The Technical Databook is revised periodically as needed to update parameters with new information or the results from new analyses It is issued as a controlled document with tracked distribution of the numbered copies

The 23 documents referenced directly by the Technical Databook (HNF-SD-SNF-TI-015) address engineering studies and analyses and characterization and testing results that have been used to support the SNF Project safety analysis process beyond the key technical data provided in the Technical Databook Some of the documents characterize the K Basins SNF with respect to mass surface area and decay heat Basic physical properties and conditions such as emissivity and climatological data are identified Other documents provide information on reaction rates that affect corrosion particulate and gas generation and water availability MCO pressures and temperatures based on the feed material process conditions and potential accident conditions also are provided

2 5 1 4 Closure of Major Technical Issues

The process hazard assessment and safety analysis processes identified key process hazards and technical issues that required resolution to ensure safe efficient implementation of SNF Project objectives The technical basis for resolution has been determined for the following topics
Zero, one, and two scrap baskets in an MCO meet safety and design basis boundaries based on updated calculations (HNF-2877)

An upper limit on free residual water has been established at 200 g per MCO (HNF-2882)

Additional methods of oxygen control, such as added gettering, fuel mixing or cladding removal, are not required to keep oxygen below the lower flammability level (HNF-3036)

The bounding value for aluminum hydroxide is 10.6 kg/MCO (HNF-3256)

The worst-case oxygen levels in an MCO will be below the lower flammability level during 40 years of storage for MCOs containing zero, one, or two scrap baskets (HNF-2876)

A limited approach to monitoring has been established that focuses on temperature pressure, and gas composition measurement of four to six MCOs for up to 2 years (HNF-3354)

Reaction rate and particulate generation assumptions are sufficiently conservative to bound observed fuel crumbling for the temperature condition that could be experienced by fuel elements during the vacuum drying process. Fuel crumbling has been observed in some tests, which raises a potential concern that this phenomena may occur in the MCO and increase the reactive surface area and the rate of heat generation during processing. The information from crumbling observed during testing does not impact the SNF Project safety basis and is bounded by conservatisms for reaction rates used in the safety analysis. Crumbling was not observed at the temperature and moisture conditions expected during MCO processing. When crumbling occurred, changes in reaction rate were not detectable within the limits of data uncertainty. The safety basis for interim storage depends on the integrity of the MCO and the limited supply of reactants rather than on reactive surface area.

Numerous scenarios have been analyzed for various combinations of cask-MCO and MCO drops during handling in the K Basins and the CSB. The results of these analyses are reported in the K Basins FSAR (WHC-SD-WM-SAR-062) and in the SNF Project FSAR annexes (HNF-3553, Annexes A and B).

2515 Margins of Safety and Conservatisms Significant design margins and safety analysis conservatisms are incorporated in the SNF Project process design and operational approach to safety. These conservatisms ensure that a robust margin of safety is provided for all phases of the process for removal processing and storage of K Basins SNF. A brief discussion of the key safety margins is provided in the paragraphs that follow.
**Particulate not Removed by Fuel Cleaning** The bounding quantities of aluminum hydroxide and uranium oxide particulate affect the pressurization of and oxygen buildup in the sealed MCO during the 40-year storage period. The quantity of fuel particulate available for release during an accident scenario also affects the radiological release calculations performed as part of the safety analysis. The significant removal of particulate during fuel washing or handling that occurs during fuel retrieval is not assumed in determining the bounding particulate and water content in an MCO. It is highly likely that fuel washing will be much more effective than is assumed by the safety basis, and this represents a substantial conservatism in the safety case. Nominal estimates of expected particulate content in an MCO quantify the probable magnitude of this conservatism—the bounding quantity is 64 kg compared to the nominal estimated value of 8 kg.

**Generated Particulate** The quantity of fuel particulate generated within an MCO after the fuel is loaded in an MCO, but before free water is removed, is conservatively calculated for the CVDF design basis accidents. The corrosion rates used for estimating the generated particulate that could be available for dispersion in certain accident scenarios is a factor of 22 higher than the nominal literature values and corrosion rates measured under prototypical atmospheric conditions by the SNF Project characterization program.

**Multi-Canister Overpack Pressurization** Chemical and radiolytic processes result in increasing pressure in a sealed MCO during interim storage at the CSB. The safety case bounding pressure expected during the 40-year storage period is <5.2 atm absolute (76.4 lb/in² absolute, or 62 lb/in² gauge). Nominal calculations of expected pressure estimate a value <16 lb/in² gauge (bounding/nominal ratio = 4.7).

**Multi-Canister Overpack Pressurization Retention Capability** With the cover cap welded in place, the MCO is estimated to have a pressurization retention capability between 1,000 to 1,300 lb/in² gauge (HNF-SD-SNF-SARR-005). This capability represents a very large margin of safety above the bounding estimated pressure of 5.2 atm absolute (76.4 lb/in² absolute). This ratio of pressure retention capability and maximum credible pressure buildup creates a bounding-to-failure margin of between 13.2 and 17.1 ([68.8 atm - 9.4 atm]/5.2 atm, or more than an order of magnitude of margin). The structural margins associated with the MCO design also provide a margin related to the potential for hydrogen burn in an MCO.

**Oxygen Concentration in a Sealed Multi-Canister Overpack** Bounding estimates of the end-of-life oxygen concentration are below a conservatively chosen lower flammability limit for all MCOs that contain one or more scrap baskets. In order to compute values that approach the hydrogen–oxygen flammability limit, the worst-case particulate and water content, fuel and scrap reactive surface area, fuel decay power, and radiolysis rate must all be assumed. Simultaneously achieving bounding safety basis values for all of these parameters in a single MCO is a significant conservatism in the safety case. HNF-SD-SNF-TI-040 documents sensitivity analyses showing the impact on the end-of-life oxygen concentrations.

**Radiological Release Calculations (Gaseous and Liquid)** Appropriately conservative parameters and calculational methods are used for SNF Project safety analyses. The degree of
conservatism and the appropriateness of assumptions used for radiological release computations were evaluated in Letter DESH 9852617 R2, *Contract Number 80232764-9-K004 — Spent Nuclear Fuel Project Safety Analysis Conservatism* (Segrest 1998)

2.5.2 Spent Nuclear Fuel Project Authorization Basis

Authorization basis is defined in DOE Order 5480.21, *Unreviewed Safety Questions* and DOE Order 5480.23, *Nuclear Safety Analysis Reports*, as follows:

*Authorization Basis* Those aspects of the facility design basis and operational requirements relied upon by DOE to authorize operation. These aspects are considered to be important to the safety of facility operations. The authorization basis is described in documents such as the facility Safety Analysis Report and other safety analyses. Hazard Classification Documents, the Technical Safety Requirements, DOE-issued safety evaluation reports, and facility-specific commitments made in order to comply with DOE Orders or policies.

This definition encompasses aspects of the facility safety basis, related technical information, technical issues, and safety analyses and other safety documentation. It also includes the design basis for the facilities. These are usually summed up in the facility FSAR and a small number of other documents. The discussion below will expand on each of these areas and arrive at the definition of the authorization basis documentation for each facility.

2.5.2.1 Spent Nuclear Fuel Project Safety Basis

Safety basis is defined in DOE Order 5480.23 as follows:

*Safety Basis* means the combination of information relating to the control of hazards at a nuclear facility (including design, engineering analyses, and administrative controls) upon which DOE depends for its conclusion that activities at the facility can be conducted safely.

The safety basis for the SNF Project's new facilities is described in this FSAR. The safety basis for the K Basins is described in WHC-SD-WM-SAR-062.

Numerous documents have been developed to support the SNF Project. The major documents that comprise the core information and analysis are summarized in this section. Figure 2-2 shows the relationship between the key facility subproject FSARs (HNF-3553, Annex A and Annex B) and the MCO Topical Report (HNF-SD-SNF-SARR-005). Figure 2-2 illustrates that the MCO Topical Report is a major reference in the safety basis of the new facilities and of the modifications to the K Basins. The MCO Topical Report is used to provide a single description of the MCO and its contents. This ensures that the different SNF Project safety analysis documents that discuss the MCO all use the same description of the MCO. The SNF Project elected to modify and update the existing K Basin FSAR (WHC-SD-WM-SAR-062) to address the canister unloading, fuel cleaning, and MCO loading operations. These were added...
to the prior functions that were already addressed by the K Basins FSAR (WHC-SD-WM-SAR-062) The CVDF and the CSB as well as the ISA, are addressed by this SNF Project FSAR and its annexes

2522 Other Safety Analysis Documents In addition to the SNF Project FSAR and its annexes several other safety documents have been developed that are important to the safety and authorization basis Most of the safety documentation was developed in concert with technical design basis documents that have been developed concurrently

The interface between the safety documentation process and the technical or design basis documents for the K Basins SNF is through the SNF Project Technical Databook (HNF-SD-SNF-TI-015) and the MCO Topical Report (HNF-SD-SNF-SARR-005) The MCO Topical Report (HNF-SD-SNF-SARR-005) is not an FSAR but is a summary document that integrates the issues design data and safety data that are common to each of the subprojects The characterization program literature reviews, engineering studies analyses, and communication documents all flow down to support the SNF Project Technical Databook and the MCO Topical Report These two documents form the foundation for the technical input for the safety documentation process

Additional safety analysis documents include the following

- The hazard analyses for each of the SNF Project FSAR Annexes and the K Basins FSAR
- The fire hazard analyses
- The criticality safety evaluation reports
- The safety analysis reports for packaging

Hazard analyses were performed and are documented separately for each of the SNF Project subprojects

In addition, an interface hazard analysis was conducted to identify and assess conditions or events occurring at one location or facility that could subsequently impact safety during cask-MCO transport or operations at another location or facility. The interface hazard analysis is documented in HNF-5245, Spent Nuclear Fuel Project Facility Interface Hazard Analysis for Production Reactor Fuel. This document assigns responsibility for analyzing new hazardous conditions if not previously identified, to the appropriate SNF Project groups or facilities to ensure development of adequate safety and operational controls using the formal DOE hazard analysis and safety analysis processes. No new hazards were identified during the interface hazard analysis that challenge the present K Basins, CVDF, and CSB FSARs, MCO Topical Report, or safety analysis report for packaging safety documentation.

Fire hazard analyses were performed and are documented separately for each of the SNF Project subprojects:

- WHC-SD-SNF-FHA-001, Fire Hazards Analysis for the K Basins Facilities at 100K Area
- HNF-SD-SNF-FHA-002, Final Fire Hazard Analysis for the Canister Storage Building
- SNF-4268, Fire Hazard Analysis for the Cold Vacuum Drying Facility
- SNF-4932, Fire Hazard Analysis for the 200 Area Interim Storage Area

Criticality safety is discussed in the SNF Project FSAR and its Annexes (HNF-3553 Annexes A, B and D), in the K Basins FSAR (WHC-SD-WM-SAR-062) and in the safety analysis reports for packaging (HNF-SD-TP-SARP-017, WHC-SD-TP-SARP-010, NAC-E-804 and WHC-SD-TP-SARP-008). Criticality events under normal and accident conditions for various MCO configurations are analyzed in HNF-SD-SNF-CSER-005, Criticality Safety Evaluation Report for the Multi-Canister Overpack. Activities related to fuel storage, fuel handling equipment, and fuel handling activities in the K Basins are evaluated in HNF-SD-SNF-CSER-010, Criticality Safety Evaluation Report for the Storage and Removal of Spent Nuclear Fuel from K Basin. Criticality evaluations for specific systems are provided in HNF-SD-SNF-CSER-011, Criticality Safety Evaluation Report for the K West Basin Integrated Water Treatment System, and in HNF-SD-SNF-CSER-006, Criticality Safety Evaluation Report for the Cold Vacuum Drying Facility's Process Water Handling System. Criticality evaluations for the handling and storage of sealed casks in the ISA are documented in several additional reports (WHC-SD-FV792-DA-004, WHC-SD-FF-CSER-002, WHC-SD-FF-CSER-003, WHC-SD-FF-CSER-004, WHC-SD-FF-CSER-006, WHC-SD-SQA-CSA-30006, WHC-SD-FF-CSER-007, HNF-3499).

November 1999
HNF-SD-TP-SARP-017 addresses the safety concerns for transport of MCOs between each of the SNF Project facilities. While FSAR Annexes A and B and HNF-SD-TP-SARP-017 are stand-alone documents, safety assumptions that are required for safe transport are established as constraints to the facility delivering an MCO and cask system for transport. The receiving areas at the CSB and CVDF have safe receiving, unloading, inspection, and storage requirements in order to meet the delivery assumptions in HNF-SD-TP-SARP-017. Other interfaces may be required in the event of a delayed shipment, such as a means of retrieving the cask for timely transport or cooling. WHC-SD-TP-SARP-010, Safety Analysis Report for Packaging (Onsite) Interim Storage Cask, NAC-E-804 Safety Analysis Report for the NFS-4/NAC-1 Spent-Fuel Shipping Cask, and WHC-SD-TP-SARP-008, Safety Analysis Report for Packaging NRF TRIGA Packaging, address the safety concerns for transport of non-defense reactor SNF to the ISA.

2 5 2 3 Design Basis Design basis is defined in DOE Order 5480.21 as follows:

**Design Basis** The set of requirements that bound the design of systems, structures, and components within the facility. These design requirements include consideration of safety, plant availability, efficiency, reliability, and maintainability. Some aspects of the design basis are important to safety, although others are not.

The design descriptions are contained in Chapter 2.0 of each SNF Project FSAR annex and of the K Basins FSAR. The discussion of how that design meets the requirements is included in Chapter 4.0 of each SNF Project FSAR annex and of the K Basins FSAR.

2 5 2 4 Authorization Basis Documentation Requirements for the authorization basis documentation are contained in DOE Order 5480.23 and include the safety analysis reports, technical safety requirements, and safety evaluation report. Because the MCO Topical Report (HNF-SD-SNF-SARR-005) contains basic design information related to the MCO and its contents and is referenced by SNF Project FSARs, it becomes part of the facility authorization basis.

During the design and construction of the new SNF Project systems, processes, and facilities, phased safety analysis documents were prepared, and reviewed and approved by DOE in order to authorize procurement of components and systems or to authorize construction. These documents were prepared in phases that supported the installation of specific equipment or the construction of defined portions of the new facilities. The safety analysis documents grew into the following safety analysis reports that contained information required by the DOE orders for preliminary safety analysis reports:

- HNF-SD-SNF-SAR-002, Safety Analysis Report for the Cold Vacuum Drying Facility Phase 2 Supporting Installation of Process Systems
- HNF-SD-SNF-RPT-004, Canister Storage Building (CSB) Safety Analysis Report Phase 3 Safety Analysis Documentation Supporting CSB Construction
The phased preliminary safety analysis reports authorize the procurement, installation, construction, and testing of the components, systems, and structures of the SNF Project's new facilities.

This SNF Project FSAR (HNF-3553) is a document that meets the content requirements and format guidelines of DOE Order 5480.23 and DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, for an FSAR and is used to authorize the final testing and operation of the SNF Project's new facilities. It contains applicable information from the phased safety analysis report documents along with new information that addresses operation of the facilities. This FSAR comprises several volumes; the first of which contains the information that applies to all of the new facilities. The other volumes (annexes) of this FSAR contain facility-specific information. Following DOE review of this FSAR and a specific facility annex, issuance of the safety evaluation report for the specific facility, the SNF Project FSAR, Volume 1, and the applicable annex volume become the primary authorization document that controls further testing, any need modification, and operations of the facility. At that time, the phased preliminary safety analysis document will no longer control activities within the facility.

Facility authorization agreements will be documented and approved. These agreements will list the documentation that establishes the facility authorization basis. It is expected that the following documents will provide the operations authorization basis:

- For the K Basins
  - K Basins FSAR (WHC-SD-WM-SAR-062)
  - K Basins Technical Safety Requirements (HNF-SD-SNF-TSR-001)
  - DOE Richland Operations Office Safety Evaluation Report
  - MCO Topical Report (HNF-SD-SNF-SARR-005)

- For the CSB
  - CSB FSAR (HNF-3553 and Annex A)
  - CSB Technical Safety Requirements (HNF-3672)
  - DOE, Richland Operations Office, Safety Evaluation Report
  - MCO Topical Report (HNF-SD-SNF-SARR-005)

- For the CVDF
  - CVDF FSAR (HNF-3553 and Annex B)
  - CVDF Technical Safety Requirements (HNF-3673)
  - DOE Richland Operations Office Safety Evaluation Report
  - MCO Topical Report (HNF-SD-SNF-SARR-005)
For the ISA

- ISA FSAR (HNF-3553 and Annex D)
- ISA Technical Safety Requirements (SNF-5047)
- DOE, Richland Operations Office Safety Evaluation Report

HNF-SD-TP-SARP-017 is the authorization basis document for transportation of loaded MOCS between SNF Project facilities. As such, there are interfacing requirements between it and facility FSARs (WHC-SD-WM-SAR-062 and HNF-3553, Annexes A and B) and the MCO Topical Report (HNF-SD-SNF-SARR-005). WHC-SD-TP-SARP-010 NAC-E-804, and WHC-SD-TP-SARP-008 are the authorization basis documents for transportation of sealed casks to the 200 East Area ISA.

2.6 CONFINEMENT SYSTEMS

For the K Basins SNF, the three major components or systems that comprise the key confinement barriers of the safety basis are (1) the MCO-transport cask system, (2) the cold vacuum drying processing system, and (3) the multi-zoned building and ventilation and filtration systems associated with the SNF Project facilities. Except during vacuum processing, the MCO provides the primary confinement barrier for the SNF Project. The MCO has various roles depending on the process function being performed. During some operations, the MCO is vented or open, while during others, it is sealed or welded closed. Table 2-3 summarizes the MCO conditions at various stages of MCO processing. For facility-specific identification and description of structures, systems, and components that perform confinement functions, see facility FSAR Annexes A and B and the MCO Topical Report (HNF-SD-SNF-SARR-005).

Each of the dry storage systems used for interim storage of non-defense reactor SNF at the 200 East ISA is described in Annex D of this FSAR.

2.7 SAFETY SUPPORT SYSTEMS

For facility-specific identification and description of the principal safety support systems, see the K Basins FSAR (WHC-SD-WM-SAR-062) and the SNF Project FSAR Annexes.

2.8 UTILITY DISTRIBUTION SYSTEMS

The Bonneville Power Administration (BPA) provides the main source of power to the Hanford Site by means of a major substation known as Midway. The BPA Midway Substation which is located near Priest Rapids Dam has three 230-kV buses. Each BPA Midway bus section is supplied by a line from the Priest Rapids generating station. BPA Midway buses No. 1 and No. 3 are also supplied by lines that are connected to the Grand Coulee 230-kV Substation and to 230-kV substations in the lower Columbia River power system. The Grand Coulee and lower...
<table>
<thead>
<tr>
<th>Process step</th>
<th>Water content</th>
<th>Fill gas</th>
<th>Expected MCO pressure (design MCO pressure)</th>
<th>Temperature</th>
<th>Boundary integrity</th>
<th>Pressure relief</th>
</tr>
</thead>
<tbody>
<tr>
<td>K Basin fuel retrieval and MCO loading</td>
<td>Submerged MCO and spent nuclear fuel</td>
<td>NA</td>
<td>--</td>
<td>20 °C</td>
<td>Open</td>
<td>NA</td>
</tr>
<tr>
<td>Transfer to CVDF</td>
<td>Wet</td>
<td>Helium in cask</td>
<td>2 lb/in² gauge initially (150 lb/in² gauge maximum)</td>
<td>&lt;38 °C</td>
<td>Vented to cask</td>
<td>MCO vented to cask and cask sealed</td>
</tr>
<tr>
<td>Cold vacuum drying (prior to draining)</td>
<td>Wet</td>
<td>Helium cover gas</td>
<td>Atmospheric</td>
<td>&lt;50 °C</td>
<td>Connected to CVDF processing systems</td>
<td>30 lb/in² gauge CVDF process rupture disk</td>
</tr>
<tr>
<td>Cold vacuum drying (vacuum processing)</td>
<td>Residual water vapor</td>
<td>Helium</td>
<td>Vacuum</td>
<td>&lt;50 °C</td>
<td>Connected to CVDF processing systems</td>
<td>30 lb/in² gauge CVDF process rupture disk</td>
</tr>
<tr>
<td>Cold vacuum drying (after residual free water removal test)</td>
<td>&lt;200 g free water</td>
<td>Helium</td>
<td>Above atmospheric pressure</td>
<td>&lt;25 °C</td>
<td>Mechanical seal</td>
<td>None</td>
</tr>
<tr>
<td>Transport to CSB</td>
<td>&lt;200 g free water</td>
<td>Helium</td>
<td>&lt;52 atm absolute (150 lb/in² gauge design)</td>
<td>&lt;38 °C</td>
<td>Mechanical seal</td>
<td>None</td>
</tr>
<tr>
<td>Transport within CSB</td>
<td>&lt;200 g free water</td>
<td>Helium</td>
<td>&lt;52 atm absolute (150 lb/in² gauge design)</td>
<td>&lt;122 °C</td>
<td>Mechanical or welded cap seal</td>
<td>None</td>
</tr>
<tr>
<td>Monitoring and MCO sealing in CSB</td>
<td>&lt;200 g free water</td>
<td>Helium</td>
<td>&lt;52 atm absolute (150 lb/in² gauge mechanical 450 lb/in² gauge welded)</td>
<td>&lt;122 °C</td>
<td>Mechanical seal or open for sampling through sample valve</td>
<td>150 lb/in² gauge sample line relief valve (active during sampling)</td>
</tr>
<tr>
<td>Interim storage at CSB</td>
<td>&lt;200 g free water</td>
<td>Helium</td>
<td>&lt;52 atm abs (450 lb/in² gauge design)</td>
<td>&lt;122 °C</td>
<td>Welded cap seal</td>
<td>None</td>
</tr>
</tbody>
</table>

CSB = Canister Storage Building  
CVDF = Cold Vacuum Drying Facility  
MCO = multi-canister overpack  
NA = not applicable
Columbia River Substations are interconnected to the BPA 500-kV transmission system. The BPA 500-kV and 230-kV transmission systems are interconnected to all the Columbia and Snake River hydroelectric generating stations and northwest steam generating plants and are also interconnected to California and British Columbia power network systems.

In addition to supplying power to the Hanford Site and other U.S. Department of Energy substations, the Hanford Site 230-kV loop normally supplies power to Ashe and White Bluffs Substations. However, if both Midway power sources are lost, sufficient power for normal operations will flow into the Hanford Site 230-kV loop from the Ashe Substation. Figure 2-5 shows the onsite transmission lines.

Power from the Ashe Substation into the Hanford Site 230-kV loop is supplied by a 115-kV line connected to the White Bluffs Substation. One 115-kV line at the White Bluffs Substation is connected to the BPA Franklin Substation. The other 115-kV line is connected to the BPA Benton Switch Substation.

Power is fed by two 230-kV lines to the 230–13.8 kV A-8 substation located near the intersection of the Yakima Barricade and 100 N highways (Route 11A and Route 4 North, respectively). The 251-W Building adjacent to the A-8 Substation contains the switchgear and control equipment. The A-8 Substation distributes 13.8 kV through overhead lines. The power is then distributed to SNF Project facilities.

For details of the facility-specific power system and components such as transformers, motor control centers, and uninterrupted power supply, see the K Basins FSAR (WHC-SD-WM-SAR-062) and the SNF Project FSAR Annexes.

2.9 AUXILIARY SYSTEMS AND SUPPORT FACILITIES

For other auxiliary and support systems in the SNF Project facilities, see the K Basins FSAR (WHC-SD-WM-SAR-062) and the SNF Project FSAR Annexes.

2.10 REFERENCES


29 CFR 1910 Occupational Safety and Health Standards *Code of Federal Regulations* as amended

ANSI/ACI 349-85 1985 *Code Requirements for Nuclear Safety Related Concrete Structures* American Concrete Institute Detroit, Michigan

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HNF-3672, 1999 *Canister Storage Building Technical Safety Requirements*, Rev 0 Fluor Daniel Hanford, Incorporated, Richland, Washington


HNF-SD-TP-SARP-017 1997 *Safety Analysis Report for Packaging (Onsite) Multi-Canister Overpack Cask* Rev 0 Fluor Daniel Hanford Incorporated, Richland, Washington


SNF-4268 1999 *Fire Hazard Analysis for the Cold Vacuum Drying Facility* Rev 0 Fluor Daniel Hanford, Incorporated Richland Washington
SNF-4820 1999 200 Area Interim Storage area Final Hazard Analysis Report, Rev 0, Fluor Daniel Hanford, Incorporated, Richland, Washington

SNF-4932, 1999, Fire Hazard Analysis for the 200 Area Interim Storage Area Rev 0 Fluor Daniel Hanford Incorporated, Richland, Washington

SNF-5047, 1999 200 Area Interim Storage Area Technical Safety Requirements, Rev 0, Fluor Daniel Hanford Incorporated, Richland Washington


WHC-SD-FF-CSER-003, 1994 Criticality Evaluation for the Long Term Storage of FFTF Fuel in Interim Storage Containers, Rev 0, Westinghouse Hanford Company, Richland Washington


WHC-SD-SNF-FHA-001 1996 Fire Hazards Analysis for the K Basins Facilities at 100K Area Rev 0 Westinghouse Hanford Company, Richland, Washington


Figure 2-1  Spent Nuclear Fuel Project Process Flow Summary
Figure 2-2  Spent Nuclear Fuel Subproject Safety Interfaces
Figure 2-3  Multi-Canister Overpack Pressurization (Bounding Safety Case)
Figure 2-4  Spent Nuclear Fuel Project Facilities and Process Physical Barriers  (Sheet 1 of 3)

Fuel Retrieval and MCO/Cask Loading

Key Physical Barriers
- Water and basin structure
- Building and ventilation system

Important Accident Considerations/Scenarios
- Dropped loads
- Criticality
- Fuel flashing/rapid underwater corrosion
- Natural phenomena and external events

Key Prevention, Mitigation and Control Features
- Cleaning process controls (affects sealed storage)
- Scrap and fuel loading procedures (affects sealed storage)
- Criticality controls
- Load lifting controls/procedures
- Basin water temperature control/heat sink
- Cask inerting and sealing procedures
- Limited fuel basket queueing (<30 days)

Key Phenomena and Analysis Results
- Particulate generation during transport
- Corrosion reaction rates
- Conditions for hydrate formation

Key Monitoring/Measurements
- Basin temperature

Transfer to CVD

Key Physical Barriers
- Sealed cask/vented MCO

Important Accident Considerations/Scenarios
- Runaway corrosion/temperature transient
- $H_2$ pressure build up
- $H_2$ deflagration
- Natural phenomena and external events

Key Phenomena and Analysis Results
- Particulate generation during transport
- Corrosion reaction rates
- Conditions for hydrate formation

Key Prevention, Mitigation and Control Features
- Limited transfer duration (<24 hours)
- Water heat sink in MCO/cask (cask annulus water level)
- Back up cooling if shipping delayed
- Transfer procedures

Key Monitoring/Measurements
- Transfer time window
- Cask pressure measured at receipt

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Figure 2-4 Spent Nuclear Fuel Project Facilities and Process Physical Barriers (Sheet 2 of 3)

Key Physical Barriers:
- MCO and vacuum purge system boundary
- Multi-zoned building and ventilation system

Important Accident Considerations/Scenarios:
- Runaway corrosion / temperature transient
- U/O_2 reaction (air ingress)
- Gas release and liquid release
- H_2 deflagration (internal and external to MCO)
- Natural phenomena & external events

Key Phenomena and Analysis Results:
- Particulate generation
- Corrosion reaction rates
- Conditions for hydrate formation
- Fuel crumbling

Key Prevention, Mitigation, and Control Features:
- CVD removal of free water (affects sealed storage)
- Inert gas purge and pressure control (SC helium)
- MCO temperature control (tempered water annulus)
- MCO and cask meeting sealing (mechanical)
- Process system and building ventilation
- Structural configuration

Key Monitoring/Measurements:
- In-process measurements
- CVD residual free water removal vacuum hold
- MCO and cask post insertion leak tests

Key Physical Barriers:
- Sealed MCO (mechanical)
- Sealed cask

Important Accident Considerations/Scenarios:
- Gas pressure build up MCO overpressure
- H_2 deflagration (within MCO)
- MCO degradation and leaking
- U/O_2 reaction (air ingress)
- H_2 deflagration in cask
- Natural phenomena and external events
- Transportation loads

Key Phenomena and Analysis Results:
- Limited particulate content
- Limited particulate water content

Key Prevention, Mitigation, and Control Features:
- Inerted MCO (pressure > atmosphere)
- O_2 gettering by uranium (no getter added)
- CVD removal of free water
- Transfer procedures

Key Monitoring/Measurements:
- Cask pressure

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Figure 2-4  Spent Nuclear Fuel Project Facilities and Process Physical Barriers (Sheet 3 of 3)

**Transport Within CSB**
- MIB
- CSB

**Monitoring and MCO Sealing in CSB**
- MIB
- Sampling/Weld Station
- CSB

**Interim Storage at CSB**
- MIB
- CSB Tube

**Key Physical Barriers**
- Sealed MCO (either welded and/or mechanical)
- Building and ventilation system

**Important Accident Considerations/Scenarios**
- Gas pressure build up, MCO overpressure
- H₂ deflagration (within MCO)
- MCO degradation and leakage
- UO₂ reaction (air ingress)
- H₂ deflagration in MIB
- Natural phenomena and external events
- Dropped loads
- Criticality

**Key Phenomena and Analysis Results**
- Limited particulate content
- Limited particulate water content

**Key Prevention, Mitigation and Control Features**
- Inserted MCO
- O₂ gettering by uranium (no getter added)
- CVD removal of free water
- Load lifting controls and procedures
- MCO pressure > atmospheric

**Key Monitoring/Measurements**
- None

**Key Physical Barriers**
- Sealed MCO (mechanical)
- Monitoring system boundary
- Building and ventilation system
- CSB tube and tube plugs
- Sampling/weld station shielding

**Important Accident Considerations/Scenarios**
- Gas pressure build up, MCO overpressure
- H₂ deflagration (within MCO)
- MCO degradation and leakage
- UO₂ reaction (air ingress)
- H₂ deflagration in MIB
- MCO overheating
- Natural phenomena and external events
- Dropped loads
- Criticality

**Key Phenomena and Analysis Results**
- Limited particulate content
- Limited particulate water content

**Key Prevention, Mitigation and Control Features**
- Inserted MCO
- O₂ gettering by uranium (no getter added)
- CVD removal of free water
- Load lifting controls and procedures
- Structural configuration
- Natural convection cooling
- Inert overpack tubes for upset impact absorbent

**Key Monitoring/Measurements**
- Building radiation
- Vault exhaust temperature
- Gas pressure indicator on limited MCOs
Figure 2-5 Hanford Site Map Showing Power Distribution System
CHAPTER 30

HAZARD AND ACCIDENT ANALYSES
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<tr>
<th>Abbreviation</th>
<th>Term</th>
</tr>
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<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
</tr>
<tr>
<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
</tr>
<tr>
<td>DBA</td>
<td>design basis accident</td>
</tr>
<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
</tr>
<tr>
<td>NRC</td>
<td>U.S. Nuclear Regulatory Commission</td>
</tr>
<tr>
<td>PHA</td>
<td>preliminary hazard analysis</td>
</tr>
<tr>
<td>SSC</td>
<td>structure system and component</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
</tr>
<tr>
<td>TSR</td>
<td>technical safety requirement</td>
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3 0 HAZARD AND ACCIDENT ANALYSES

3 1 INTRODUCTION

The purpose of this chapter is to provide an overview of the process used to systematically identify and assess hazards to evaluate the potential internal, external, and natural phenomena events that can cause the identified hazards to develop into accidents. The results of this hazard identification and assessment process specific to individual Spent Nuclear Fuel (SNF) Project facilities are provided in the facility annexes to the SNF Project Final Safety Analysis Report (FSAR). Hazard analysis considers the complete spectrum of accidents that may occur because of facility operations, analyzes potential accident consequences to the public and workers, estimates likelihood of occurrence, identifies and assesses associated preventive and mitigative features, identifies potential safety structures, systems, and components (SSCs), and identifies a selected subset of accidents designated design basis accidents (DBAs), to be formally defined in accident analyses. Subsequent accident analyses evaluate these DBAs for comparison with release limits and evaluation guidelines to identify and assess the adequacy of all safety SSCs. These analyses pave the way for the development of the technical safety requirements (TSRs) and the needed emergency response plans.

When the FSAR is approved by the U.S. Department of Energy (DOE), it will establish the authorization basis for the SNF Project. Changes to the facility during operations will be reviewed to ensure they do not affect the authorization basis. This review process is described in Chapter 17.0 and is termed the unreviewed safety question process. When activities or projects result in a positive unreviewed safety question evaluation, an authorization basis change package will be developed. This package will include text changes for the SNF Project FSAR, the appropriate FSAR Annexes, and the TSR documents (as required). The changes will be for DOE review and approval.

This chapter describes the methodology for and approach to hazard and accident analyses. The facility FSAR Annexes cover hazard identification, facility hazard classification, hazard evaluation, and accident analyses. Products of the FSAR Annexes include the following:

- Identification of hazardous materials and energy sources present by type, quantity, form, and location
- Facility hazard classification, including segmentation in accordance with DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23 Nuclear Safety Analysis Reports*
- Identification in the hazard analysis of the spectrum of potential accidents at the facility in terms of largely qualitative consequence and frequency estimates and a summary of this activity that includes
  - Identification of planned design and operational safety improvements
Summary of defense in depth and other items needing TSR coverage in accordance with DOE Order 5480.22 *Technical Safety Requirements*

Summary of the significant worker safety features including identification of safety-significant SSCs for worker safety and any relevant programs to be covered under TSR administrative controls

Summary of design and operational features that reduce the potential for large material releases to the environment

Identification of the limited set of unique and representative accidents (i.e., DBAs) to be assessed further in accident analyses

Accident analyses of DBAs identified in the hazard analysis and a summary of this activity that includes for each accident analyzed the following:

- Estimation of source term and consequence
- Documentation of the rationale for binning frequency of occurrence in a broad range in hazard analysis (detailed probability calculations not required)
- Documentation of accident assumptions and identification of safety-class and safety-significant SSCs based on evaluation guidelines

Several major accident types were considered in the SNF Project hazard and accident analyses. Project safety basis accidents considered include the following:

- Multi-canister overpack (MCO) overpressurization or leakage
- Rapid corrosion or temperature transient (H₂O–uranium)
- Uranium metal or uranium hydride burn (O₂–uranium)
- Radiological releases during processing and treatment
- Hydrogen combustion or explosion
- Nuclear criticality
- Dropped loads that cause cold vacuum drying system breach (radiological release)
- Transport accidents (HNF-SD-TP-SARP-017)
- MCO or fuel overheating
- Natural phenomena and external events (e.g., fire)
- MCO mechanical breach or significant geometry changes caused by shear drop, or collision impact forces

Based on a comprehensive hazard analysis process, representative DBAs are selected for in-depth analysis. The DBA represents a bounding case for a category of events. The analysis process results in identification of safety-class and safety-significant SSCs and the functional requirements needed to prevent or mitigate potential accident sequences and to ensure that the facilities can be operated in a safe, controlled manner. The analysis also identifies a set of controls (i.e., TSRs) that ensure all identified vulnerabilities associated with the operation of the facility have been adequately addressed. The comprehensive hazard and safety analyses results for each phase of the SNF Project are documented, with preventive or mitigative features, in the facility FSAR Annexes.

The SNF Project product specification (HNF-SD-SNF-OCD-001) provides the limiting values of all key parameters. Based on limiting values assumed in the hazard and DBA analyses, the SNF Project has interface requirements among its facilities to ensure that where the different facilities rely on certain functional performance requirements to be satisfied, those performance requirements will be met. For example, the Cold Vacuum Drying Facility (CVDF) assumes that the MCO is loaded with at most two scrap baskets, where scrap baskets may only be loaded at the top and bottom of the MCO. The Canister Storage Building (CSB) requires that an MCO delivered from the CVDF to the CSB contain less than 200 g of free water and confine combustible hydrogen gases within the mechanically sealed MCOs.

### 3.2 REQUIREMENTS

The requirements that form the basis for performing hazard and accident analyses to establish the safety basis of SNF Project facilities are found in HNF-SD-SNF-RD-001, *Spent Nuclear Fuel Project Standards/Requirements Identification Document*. Specific requirements applicable for this chapter include the following:

- DOE Order 5480 22, *Technical Safety Requirements*
- DOE Order 5480 23, *Nuclear Safety Analysis Reports*
- DOE Order 6430 1A, *General Design Criteria*


In Letter 95-SFD-167, *Implementation of the K Basins Spent Nuclear Fuel Project (SNFP) Regulatory Policy* (Sellers 1995), DOE established the requirement for new SNF Project
facilities to achieve "nuclear safety equivalency" to comparable U S Nuclear Regulatory Commission (NRC)-licensed facilities. The SNF Project identified the NRC requirements that were needed in addition to existing and applicable DOE requirements to establish nuclear safety equivalency. These NRC requirements and the process used to identify them are documented in HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward Additional NRC Requirements, WHC-SD-SNF-DB-009, Canister Storage Building Natural Phenomena Hazards and WHC-SD-SNF-DB-010, Cold Vacuum Drying System Natural Phenomena Hazards. These documents establish design requirements to be met for the new SNF Project facilities to achieve this equivalency.

The following NRC rule and guidance were considered in the development of HNF-SD-SNF-DB-003, WHC-SD-SNF-DB-009 and WHC-SD-SNF-DB-010 and have particular significance to Chapter 3. However they do not by themselves establish requirements for the facilities or the Chapter 3 accident analyses:

- Title 10 Code of Federal Regulations, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR 72)
- NRC Regulatory Guide 3 26 Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants
- NRC Regulatory Guide 3 48 Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)

3.3 HAZARD ANALYSIS

Hazard identification and evaluation can provide a thorough, predominantly qualitative evaluation of the spectrum of risks to the public, workers and the environment from accidents involving the hazards identified in the analysis. The hazard analysis determines the material system MCO handling, processing, storage and facility characteristics that can produce undesirable consequences, followed by the assessment of hazardous situations associated with a process or activity. Largely qualitative techniques are used to pinpoint weaknesses in design or operation of the facility (MCO processing, handling, or storage) that could lead to accidents. The hazard analysis examines the complete spectrum of potential accidents that could expose members of the public, onsite workers, facility workers and the environment to hazardous materials. A selection process is used to identify accidents with the potential to cause unacceptable consequences. These accidents are further analyzed, and a set of controls is selected and implemented to ensure that the facility can be designed, constructed, operated, and decommissioned safely and to meet DOE requirements. This section presents the methodology used to perform the SNF Project hazard analysis and to identify the resulting candidate accidents selected for more comprehensive analyses in subsections of Section 3.4.2 of the facility FSAR Annexes.
Hazard analyses were prepared to support the phased safety analysis reports. Final hazard analyses were performed to support the accident analyses by providing a basis for selecting representative and bounding DBAs. The final hazard analyses systematically reviewed the final designs to identify any additional hazardous materials or energy sources that have the potential to initiate an accident that may require further review or analysis. A specific and comprehensive analysis of all fire hazards associated with the facilities have been completed and augment the standard hazard analysis.

The revised hazard analyses were prepared using baseline design information identified in Chapters 20 of the facility FSAR Annexes, system design descriptions, and other design requirements documents, including relevant design drawings, design calculations, supporting documents, and operational plans.

3.3.1 Methodology

This section describes the methodology used to identify and evaluate the SNF Project facilities hazards. The hazard evaluation process identifies hazardous conditions, determines causes and preventive and mitigative features, and qualitatively estimates the consequences and frequencies of occurrence. The results of the application of this methodology are presented in the facility FSAR Annexes. The hazard analysis was based on American Institute of Chemical Engineers methodology that conforms with the guidance provided by DOE-STD-3009-94 which implements the requirements of DOE Order 5480.23.

3.3.1.1 Hazard Identification

The hazard analyses considered events occurring during normal operations at the facilities to ensure that there were no adverse consequences for the public or the workforce or contamination of the environment. The hazard analysis identified all conditions that could occur during normal operations and during abnormal and accident conditions at the facility. The hazard analysis therefore focused on hazardous conditions created by equipment failures, errors, or other accidents leading to release of hazardous material in the facility.

A hazard is defined in DOE Order 5480.23 as a source of danger (i.e., material energy source or operation) with the potential to cause illness, injury, or death to personnel or damage to an operation or to the environment (without regard for the likelihood or credibility of accident scenarios or consequence mitigation). Hazard identification for the SNF Project facilities was based on examination of the major building areas in the facility designs. The processes and activities that can take place within each area were identified. A hazardous material/energy source checklist, provided in Table 3-1, was based on a standard table and modified to include facility-specific concerns. This table was used to group potentially hazardous materials and energy sources as they were identified in each of the major building areas. An alpha-numeric designator was assigned to each hazardous material and energy source identified. The designator combined the acronym for the building area, the designation letter for the general hazard group, and the hazard number for the specific type cause, or source. For example, AA-A-01 identifies an electrical source in the administrative area, specifically battery banks. If a single hazard could result in more than one consequence, different features may be needed to prevent or mitigate each.
<table>
<thead>
<tr>
<th>Location</th>
<th>Y N</th>
<th>A Electrical</th>
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consequence. Such possibilities, discussed in separate rows in the hazard evaluation table, are distinguished by appending a lowercase letter to the final hazard number (e.g., AA-A-01a, AA-A-01b, and AA-A-01c).

Identified hazards were grouped by their general form (e.g., electrical, thermal, friction) and by specific type, cause, or source (e.g., motors, power tools, wiring). The hazards were further identified by determining hazardous conditions resulting from the hazardous material or energy source, establishing accident initiators or causes, and defining resultant accident sequences.

The hazardous materials and energy sources with the potential to cause an uncontrolled release of radioactive or other hazardous material were screened from those that present only standard industrial hazards. Standard industrial hazards are defined in DOE-STD-3009-94 as those hazards that “are routinely encountered in general industry and construction and for which national consensus codes and/or standards (e.g., Occupational Safety and Health Administration, transportation safety) exist to guide safe design and operation without the need for special analysis to define safe design and/or operational parameters.” These standard industrial hazards were dismissed from further analysis in the facility FSAR Annexes. Standard industrial hazards that can contribute to the uncontrolled release of radioactive or hazardous material, and the remaining hazards from the hazardous material/energy source checklist that were not dismissed represent hazards requiring further characterization and evaluation as potential accident analysis candidates.

### 3.3.1.2 Hazard Evaluation

The hazard evaluation was a structured and systematic examination of the SNF Project facilities and operations described in Chapter 2.0 of each facility FSAR Annex. Standard industry (American Institute of Chemical Engineers) hazard evaluation techniques were used. The hazard evaluation was performed by a team of cognizant SNF Project operations and engineering personnel, safety analysts familiar with the SNF Project, and technical experts in specialty areas.

A preliminary hazard analysis (PHA) technique is used to analyze facility operations and situations that are primarily operations driven. A PHA focuses in a general way on the hazardous materials and major process areas of a plant. The PHA technique normally is used in the preliminary phase of plant development for cases where experience provides little or no insight into potential safety problems. The technique also may be helpful when analyzing large existing facilities or prioritizing hazards when circumstances prevent a more extensive technique from being used.

In the PHA, lists of hazards and generic hazardous situations are examined to determine whether any apply to the facility being analyzed using the following process characteristics:

- Raw materials
- Intermediate and final products (and their reactivity)
- Plant equipment
- Facility layout
- Operating environment
- Operational activities (e.g., testing, maintenance)
- Interfaces among system components
- Energy sources

A PHA yields a qualitative description of the hazards related to a process design. A PHA also provides a qualitative ranking of hazardous situations that can be used to prioritize recommendations for reducing or eliminating hazards in subsequent phases of the life cycle of the process. Using the PHA technique requires that analysts have access to available plant design criteria, equipment specifications, material specifications, and other sources of information.

Each hazard evaluation was performed using a table to record the specific building location and hazard as described in Section 3.3.1.1 and the evaluation factors identified in this section. Each hazardous condition was reviewed to determine causes for the hazard, potential accidents that could result from the presence of each hazard, and consequences to the public offsite, collocated and facility workers onsite, the environment, or the specific facility. Safety features, segregated into engineering and administrative features, were identified for each hazard based on preventing or mitigating the consequences. Qualitative estimates of the frequency and consequences of the hazardous condition were assigned.

The hazard evaluation included the following information:

- Hazardous condition
- Cause
- Frequency
- Consequence

Potential hazardous conditions (excluding those identified as common industrial hazards) are developed based on the hazards identified and the application of the hazard evaluation methods described above. The description of each hazardous condition includes release of material from a location as the result of an event.

The hazard evaluation tables contain descriptions of causes for hazardous conditions. The identification of causes was based on the hazard identification results, reviews of the systems and historical event data, and other sources of information (existing safety documentation), including discussions with experienced facility personnel. The causes of hazardous conditions include internal events, external events, and natural phenomena events. Single and multiple failures (equipment and human errors) were considered as well as common-cause failures. Both human errors of commission and omission (i.e., human reliability) have been identified as potential causes or contributors to hazardous conditions. The potential causes of a postulated hazardous condition are identified to support a qualitative frequency evaluation.
The assessment of the consequence for each hazardous condition was a qualitative judgment. The assessment took into consideration the impact of passive features (e.g., structures, barriers) listed in the hazard analysis but not the impact of active features or planned controls (e.g., valves, shipping restrictions). The qualitative criteria for consequence assessments are as follows:

**S3**
On the basis of material at risk and causes postulated, there is sufficient material and release energy to affect a receptor at the nearest point of uncontrolled public access.

**S2**
On the basis of material at risk and causes postulated, there is sufficient material and energy to affect an onsite receptor (collocated worker) 100 m from the source of the release.

**S1**
On the basis of material at risk and causes postulated, the release is confined to the facility and affects facility workers.

**S0**
On the basis of material at risk and causes postulated, there is insufficient material released to affect facility workers.

The more severe consequence categories encompass the less severe consequence categories. For example, a hazardous condition assessed as having onsite consequences (S2) is also considered to have facility worker consequences (S1).

The assessment of frequency also was a qualitative judgment. The assessment took into consideration the impact of designed passive features (e.g., structures, barriers) but not the impact of active features or planned controls (e.g., valves, shipping restrictions). The frequency assessments were based on initiating event frequencies of a per year basis. The qualitative criteria for frequency assessments are as follows:

**F3**
The hazardous condition based on the causes postulated is anticipated to occur during the facility's lifetime ($10^2$ to $10^1$ per year).

**F2**
The hazardous condition based on the causes postulated is foreseeable, but unlikely ($10^{-4}$ to $10^{-2}$ per year).

**F1**
The hazardous condition based on the causes postulated is perhaps possible but extremely unlikely ($10^{-6}$ to $10^{-4}$ per year).

**F0**
The hazardous condition based on the causes postulated is considered too improbable ($<10^{-4}$ per year) to warrant further consideration.
3.3.2 Hazard Analysis Results

The results of the hazard analysis were used in the selection of DBAs. Analysis of the unmitigated consequences of the DBAs was used to identify safety function SSCs and to determine their classification. Events identified by the hazard analysis as having significant consequences to offsite and onsite receptors were assigned to risk bins organized by major building area. Figure 3-1 (which is based on Figure 3-3 in DOE-STD-3009-94) presents the likelihood and consequence ranking matrix. In terms of the risk binning process, the accidents chosen from the hazard analysis for further analysis as DBAs were all events identified in consequence categories S3 and S2. Events identified in frequency category F3 as having S1 or S2 consequences are not ignored, they are reviewed and addressed as abnormal events as described in Section 3.3.3 'Methodology for Abnormal Event Identification, Analysis, and Documentation'. These events also are discussed in Section 3.3.3, "Worker Safety." Consistent with DOE-STD-3009-94, the events located in risk bins represent "situations of concern," or "situations of major concern." These events were evaluated as candidate DBAs. The final list of candidate DBAs, sorted by risk ranking, formed the basis for the selection of DBAs analyzed in Sections 3.4 of the facility FSAR Annexes. DBA selection is addressed in Section 3.3.2.5.

The design SSC functions and facility controls used in the final hazard analysis were checked against the up-to-date specific facility designs described in Chapters 2.0, 4.0, and 5.0 in the facility FSAR Annexes.

3.3.2.1 Hazard Identification

Completed facility-specific hazard analysis tables are shown in Sections 3.3.2.1 of the facility FSAR Annexes. As described in Section 3.3.1.1, the hazards were identified by form (e.g., electrical, thermal, friction), type (e.g., motors, power tools, wiring), and building location. The radionuclide content of the special nuclear material is the main inventory of hazardous material in the K Basins, CVDF, and CSB. The toxicological hazards of the radionuclide inventory were reviewed. As described in Section 3.4.1.1, the radiological guidelines are more limiting than the toxicological guidelines for the release of SNF particulate. The radioactive particulate on SNF is primarily oxides of uranium which are not expected to change under current accident conditions. Other hazardous material identified by the hazard identification process includes pyrophoric metals and hydrides, oxidizers, hydrogen, diesel fuel, and other flammable or combustible materials. The specific characteristics of these hazards are identified in detail in the hazard analysis as documented in the facility FSAR Annexes. Specific and comprehensive analyses of fire hazards associated with SNF Project facilities have been completed to augment the standard hazard analyses.

The CVDF and the CSB do not have operating histories, so major hazards resulting from facility operations cannot be identified or summarized as suggested by DOE-STD-3009-94. K Basins operating history is limited to movements of fuel canisters with manual canister grappling tools that have been factored into the K Basins hazard analysis. However, the spent fuel handling and storage activities are similar to those of independent spent fuel storage installations issued materials licenses under 10 CFR 72. Similarity exists in that these independent spent fuel storage installations must handle SNF containers similar to the SNF Project's.
transportation casks and MCOs. The 10 CFR 72 dockets, including notices of violations issued and reported under 10 CFR 72.75, “Reporting Requirements for Specific Events and Conditions” were reviewed. The facilities reviewed included Fort St. Vrain, H.B. Robinson, Pacific Sierra Nuclear, Vectra-Pacific Nuclear Fuel, Oconee, Prairie Island, Calvert Cliffs, Surry, Point Beach and several facilities licensed under 10 CFR 72. Major hazards from similar independent spent fuel storage installations were considered when performing the hazard analyses for the SNF Project facilities. These hazards include generation of combustible gases, failure of confinement barriers, defects in cask integrity, and the spread of external contamination.

3.3.2.2 Final Hazard Classification. In accordance with DOE-STD-1027-92, the final hazard categorization for each of the SNF Project facilities was performed based on the facility’s final hazard analysis and accident analyses. The final categorizations are based on the unmitigated releases of materials at risk causing the DBAs for the specific facility as described in Sections 3.4 of the facility FSAR Annexes. The facility material inventories were compared against the DOE-STD-1027-92 threshold quantities and final facility hazard categorizations were determined as described in Sections 3.3.2.2 of the facility FSAR Annexes.

3.3.2.3 Hazard Evaluation. The final facility hazard analyses characterized hazards not considered to be standard industrial hazards in the context of the actual SNF processing, handling, and storage to be performed in SNF Project facilities. For example, while an MCO is being handled and stored in a storage tube, a number of initiating events may put the MCO radionuclide inventory at risk. These initiating events were listed and the characterization information identified for the energy sources related to the initiating event. The characterized hazards were entered in the hazard analysis tables by facility area. Each hazardous condition was then identified and recorded along with the potential accidents arising from the presence of the hazardous condition, potential causes and consequences of the hazardous condition, existing design features to prevent the condition or mitigate the consequences, administrative features planned to prevent the hazardous condition or mitigate the consequences, and qualitative estimates of the frequency and consequences of the hazardous condition.

The hazard groupings described in Section 3.3.1.1 provide for systematic evaluation of the hazards based upon common release phenomena. The qualitative severity of the hazard consequence categories described in Section 3.3.1.2 considers the potential impact of the hazard on the offsite public, collocated onsite worker, and the facility worker. The hazard analysis also examined the potential for environmental contamination and listed those preventive or mitigative features that are already a part of the design.

3.3.2.3.1 Planned Design and Operational Safety Improvements. According to DOE-STD-3009-94, this section discusses commitments (resulting from the hazard evaluation) for the planned major design fuel handling and cleaning, MCO loading, MCO handling, or MCO storage improvements for the specific facility that are not yet implemented. In general, because the facility design and accident analyses were developed in parallel, the opportunity to provide feedback for design consideration has been available. As the hazard evaluation and specific accident analyses have progressed for each of the SNF Project facilities, cost-effective modifications that improve safety have been incorporated into the design. Therefore, in general...
there are no outstanding major improvements for improved safety planned as a result of the hazard evaluation that are not part of the current design and planned facility operations (MCO processing, handling or storage) at the SNF Project facilities.

### 3.3.2.2 Defense in Depth

The SNF Project is committed to protecting the health and safety of the environment, workers, and the public through its commitment to and implementation of numerous environmental, safety, and health programs. These activities are identified and discussed in the programmatic chapters of the FSAR. In particular, HNF-3552, *Spent Nuclear Fuel Project Execution Plan*, describes these efforts to ensure a “defense-in-depth” approach to project implementation. A summary of fundamental points relevant to the concept of defense in depth is as follows:

- **Builds layers of defense** against releases of hazardous materials (radiological and toxic chemical) to reduce reliance on any one layer.

- **Can be considered independent of functional classification of feature** (i.e., core features are expected to be safety class and safety significant as required, but not all defense-in-depth features are required to be safety class or safety significant).

- **Includes barriers to hazardous releases and protection of those barriers**
  - Barriers can be physical (design features) and operational (administrative controls).
  - Barriers depend upon high-quality design and construction and competent operating personnel.
  - Innermost layers of defense are preferably highest quality, highest reliability barriers.

- **Provides protective functions that are some combination of passive design features, automatic design features, design features required to alert operators to action, and programmatic features** (e.g., procedures, training, and TSRs).

- **Outer layers of defense (barriers) may provide**
  - Diverse control of a hazard already controlled by inner barriers.
  - Additional hazard mitigation should inner barriers fail.
  - Protection of the inner barriers to prevent their failure.

All SSCs are designed in accordance with applicable codes and standards with a high degree of reliability and simplicity. The designs also encompass human factors considerations to ensure that operations can be conducted safely.
Safety-Significant Structures, Systems, and Components  This section in the facility FSAR Annexes identifies defense-in-depth SSCs for the SNF Project facilities as safety-significant SSCs. Safety-significant SSCs are predominantly required to prevent or mitigate consequences of postulated accident events to the collocated onsite worker. In addition, DOE-STD-3009-94 suggests that SSCs be designated as safety significant if they play a key role in defense in depth (or worker safety). The severity of the event being prevented or mitigated and the number of barriers present are provided in the standard as guidance for the identification of defense-in-depth safety-significant SSCs.

Technical Safety Requirements  This section in the facility FSAR Annexes identifies defense-in-depth SSCs requiring TSR coverage. All identified TSRs have been chosen in relation to postulated accident events that would challenge accident consequence evaluation guidelines for offsite public and collocated onsite worker receptors. The identified TSRs are indicated in the DBA sections and are further explained in Chapters 5 of the facility FSAR Annexes. In addition, criticality prevention features are controlled by TSRs as identified in Chapters 5 of the facility FSAR Annexes.

3.3.2.3.3 Worker Safety  This section in the facility FSAR Annexes identifies significant worker safety-related SSCs and TSRs based solely on worker safety considerations. Worker safety for the SNF Project facilities is ensured by a combination of design features that reduce exposure to radioactive, toxic, and industrial hazards and by institutional practices that, in total, provide adequate protection of workers from these hazards. Protection of the facility worker from standard industrial hazards identified for the SNF Project facilities is achieved through adherence to the institutional safety programs described in this safety analysis report and documented in lower-tier documents such as health and safety plans and job hazard analyses. Such industrial hazards do not require specific safety-significant SSCs or TSR-level administrative features. Therefore, in accordance with the guidance of DOE-STD-3009-94, the remainder of this section deals with protecting workers from the hazards of facility operation exclusive of standard industrial hazards.

The hazard analyses in the facility FSAR Annexes provide an overview of the major features protecting facility workers at their respective facilities. Worker safety features are an integral part of facility design and operation. The major features of worker protection are identified in tables in the facility FSAR Annexes and are categorized by hazard. The hazard location and the potential accident and consequence are identified along with protective features including passive, active, and administrative features. As discussed in the facility FSAR Annexes, the controls identified in the analysis of the DBAs in conjunction with the safety features identified in the hazard analysis and the institutional programs, are demonstrated to be adequate to ensure worker safety.

Protection Against Releases  The hazard analysis was used to develop a list of anticipated conditions that could result in releases affecting only facility workers (i.e., hazardous conditions identified as frequency category F3 and consequence category S1 and having no controls). These hazardous conditions are presented in the individual facility FSAR Annexes and November 1999.
were reviewed to determine whether they represented potential for serious injury or death to the worker. Any such conditions were determined to be safety significant for worker safety.

33234 Environmental Protection This section of the facility FSAR Annexes identifies the external hazards and the means for their prevention and mitigation. The external hazards associated with each facility’s operation all involve the potential release of contaminants. The release pathway for contaminants is only via the air to the boundaries and receptors discussed in Sections 1313 of the facility FSAR Annexes. Liquid release hazards or accidents, such as contaminant releases to the ground or groundwater, are the key hazards to be considered in protecting the environment and are identified in the facility FSAR Annexes. In general, in SNF Project facilities where liquid releases are possible, they are contained by design of the facility. However, the potential exists for the release of large quantities of contaminated water from the K Basins to the environment. Potential environmental consequences, including offsite releases, and required prevention and mitigation features are discussed in subsections of Section 342 of the facility FSAR Annexes. Implementation of the prevention and mitigation features will prevent large releases that could have significant environmental impact.

In general, the SNF Project facilities minimize or do not use toxic chemicals during operations. However, the toxicological hazards of the radionuclide inventory were reviewed, as described in Section 3411, and it was shown that for all accidents involving the release of spent fuel particles and for all frequency ranges, the radiological guidelines are more limiting than the toxicological guidelines (i.e., SSCs and controls necessary to limit radiological risk to within guidelines are more than adequate to limit toxicological risk to within toxicological limits) (HNF-SD-SNF-TI-059).

In general, the SNF Project facility features that protect the onsite collocated worker and public against radiological exposure also serve to prevent and mitigate radiological release to the environment. In addition, sitewide programs for environmental monitoring provide for assessment of the impact of facility releases. Normal MCO processing, handling, and storage activities are expected to have a minor impact on the local and regional environment.

33235 Accident Selection This section of the facility FSAR Annexes provides the results of the accident selection process based on the hazard analysis process. The methodology for the selection of DBAs is specified by DOE Order 5480-23, which states that DBAs are classes of accidents that are used to provide the design basis parameters for release barriers and mitigating systems. As such, this is a common requirement among the SNF Project facilities and is described below.

The representative and bounding conditions in turn define an envelope of operations that is used to evaluate safe operation of the facility, both in terms of its initial startup and operating lifetime and during the evaluation of facility modifications using the unreviewed safety question determination process. The basic accident selection methodology used for SNF Project facilities and documented in the facility FSAR Annexes is as follows.
Use the hazard analysis to identify hazards and postulate accident scenarios for each operational area of the facility (hazards that have the possibility of affecting other operational areas of the facility are either evaluated within the operational area in which they originate or are referred to the other area for further evaluation).

- Make preliminary evaluations of frequencies and consequences for these events.
- Organize the events having significant consequences into categories.
- Choose a bounding event representing the most severe consequence and highest risk scenario, from each of these categories for further analysis as a DBA.
- Consider also the less severe events that pose unique and important phenomenological challenges to the facility.

Candidate accidents are selected by binning the identified hazards in terms of frequency, consequence, and risk. The frequency and consequence categories that are used in this binning process are defined in Section 3.3.1.2. The final set of candidate accidents resulting from the hazard binning process is presented in Section 3.3.2.3.5 of the facility FSAR Annexes and contains all identified category S3 and S2 events. All of these events and the controls selected for their prevention and mitigation, are described in DBA discussions in subsections of Section 3.4.2 of the facility FSAR Annexes. The selected DBAs then establish the functional requirements that the safety-class and safety-significant SSCs must meet.

3.3.3 Methodology for Abnormal Event Identification, Analysis, and Documentation

The DOE orders are the controlling requirements for the SNF Project; however, additional NRC criteria were established in the Concurrence with the K-Basins Spent Nuclear Fuel Project Policy on Nuclear Safety Requirements (Grumbly 1995) (hereafter referred to as the Policy). The Policy required new SNF Project facilities to achieve “nuclear safety equivalency” to comparable NRC-licensed facilities. NRC equivalency requirements were identified in WHC-SD-SNF-DB-002 Spent Nuclear Fuel Project Path Forward Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities and HNF-SD-SNF-DB-003.

Specific NRC equivalency requirements of interest are those associated with facility conditions not analyzed as “normal operations or accidents” as required by the DOE safety analysis process. These conditions, described as “off-normal operations or abnormal operations” in the NRC documentation, are expected to occur during the lifetime of the facility but do not result in radiological releases outside the facility. Facility worker exposure to ionizing radiation while responding to abnormal events will be controlled by site and facility radiological protection and ALARA (as low as reasonably achievable) requirements.

This section describes the process used to identify, analyze, and document abnormal events fitting the NRC profile for “off-normal or abnormal operations.” This requirement for inclusion
of abnormal operations is part of the NRC equivalency requirements identified in HNF-SD-SNF-DB-003. The process builds upon the discussion in Section 3.3.1.2 and is diagramed in Figure 3-2.

This section also discusses implementation of NRC nuclear safety equivalency ANSI/ANS-57 9-1992 *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)* and ANSI/ANS-57 2-1983, *Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants*, present a set of events for consideration in the design and accident analyses for a spent fuel storage installation. The discussion in this section uses the terminology of ANSI/ANS-57 9-1992 but the terminology of ANSI/ANS-57 2-1983 also applies. Table 3-2 defines the event frequency that is related to each of the events outlined in these standards and relates the terminology used in ANSI/ANS-57 9-1992 to the terminology used in ANSI/ANS-57 2-1983.

<table>
<thead>
<tr>
<th></th>
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</thead>
<tbody>
<tr>
<td>Design event I</td>
<td>Plant condition I</td>
<td>Expected to occur regularly or frequently in the course of normal operation</td>
</tr>
<tr>
<td>Design event II</td>
<td>Plant condition II</td>
<td>Expected to occur with moderate frequency or on the order of once during a calendar year of operation</td>
</tr>
<tr>
<td>Design event III</td>
<td>Plant condition III</td>
<td>Infrequent event that would reasonably be expected to occur during the lifetime of the installation</td>
</tr>
<tr>
<td>Design event IV</td>
<td>Plant condition IV</td>
<td>Postulated because the consequences of such an event may result in the maximum potential impact on the immediate environs</td>
</tr>
</tbody>
</table>

Note: Events in this table are used in the implementation of U.S. Nuclear Regulatory Commission nuclear safety equivalency.

*ANSI/ANS 57 9 1992 1992 Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)*
American Nuclear Society La Grange Park Illinois

This section is limited to consideration of abnormal and accident conditions. Design event I occurrences are associated with normal operation. Protection of the facility worker from normal and abnormal events will be addressed in Chapters 7 and 8 of the facility FSAR Annexes as suggested by DOE-STD-3009-94. Chapter 7 also contains a summary of the calculations performed to determine the impact of normal operations upon members of the public. Facility loadings experienced during design event I conditions (i.e., normal operation) such as dead and live weight, pressure, and thermal loading will be discussed in Chapters 2 and 4 of the facility FSAR Annexes as suggested by DOE-STD-3009-94. Similar loadings for the MCO are discussed in HNF-SD-SNF-SARR-005, *Multi-Canister Overpack Topical Report*. Design event II, III, and IV occurrences are identified by the hazard identification process discussed in Section 3.3.1. Facility-specific consequence analyses for design event II, III, and IV hazards are performed as discussed in Section 3.3.3 of the facility FSAR Annexes.

Design events II and III are associated with abnormal events. Abnormal events are operating conditions resulting from situations outside of normal operations, where normal operations are defined by process flow diagrams, system design descriptions, and operation and maintenance procedures. This category encompasses abnormal, upset, and off-normal operations from malfunctions of systems, operating conditions, or operator errors. This category applies to events that are expected to occur annually or several times during the lifetime of the facility. Expected consequences of abnormal events can include an increase in occupational exposure of workers or guidelines.

Design events IV are associated with accident occurrences. Accidents are addressed in Section 3.4 of the SNF Project FSAR.

### 3.3.1 Use of Facility Hazard Analysis Results

Consistent with the DOE Order 5480.23 and DOE-STD-3009-94 qualitative likelihood and severity classifications, frequency and consequence rankings were assigned to each hazardous condition during the hazard analysis process. The frequency and consequence rankings are described in Section 3.3.2.

The frequency/consequence matrix from DOE-STD-3009-94 and the definition from Section 3.3.2 were used to identify events fitting the NRC off-normal or abnormal operations profile. Hazard analysis tables, containing the frequency and consequence rankings for hazardous conditions and potential accidents or events, are included in SNF Project facility-specific hazard analysis report.

Hazardous conditions and their associated potential accidents or events that fall into boxes identified as F3 and part of F2 (frequencies from annually to once per 100 years) and S0 or S1 are likely to occur during the facility design lifetime. SNF Project facility design lifetimes are 40 years for the CSB and less than 5 years for the CVDF and the K Basins. These events have been analyzed in the category of abnormal events. The specific identification and analysis of these events is further described in the SNF Project facility-specific FSAR Annexes.

### 3.3.2 Abnormal Events Binning

Abnormal events identified by the above process were binned into groups having similar operational impact on the facility. While there are numerous...
ways to bin events the SNF Project determined that grouping events according to their operational impact provided a logical connection to operational-related procedures and the steps necessary to return the facility and process to normal operations. For each bin the following format and descriptive information is provided:

**Event** The event, or events, having similar operational impacts including the location of event type of failure or mis-operation and system or systems involved

**Postulated Cause of Event** The occurrences that could initiate the event or events under consideration

**Detection of Event** The means or methods (e.g., visual or audible alarms or routine inspections) to detect events

**Analysis of Effects and Consequences** The consequences of the event, or events, under consideration

**Corrective Actions** Corrective actions necessary to return to a normal situation

Chapter 7.0 of this volume and the facility FSAR Annexes contain the results of calculations of the impact of abnormal events on members of the public.

### 3.4 ACCIDENT ANALYSIS

This section presents the methodology for formal development and analysis of the potential accidents at the SNF Project facilities. Results and development of preventive and mitigative features are provided in the facility FSAR Annexes.

#### 3.4.1 Methodology

This section presents the methodology used to develop the potential accidents described in the facility FSAR Annexes. The accident analysis for each DBA starts with a description of the accident scenario with the major assumptions identified. The accident source term is then determined. Source terms for the accidents have been obtained through phenomenological and system response calculations. Once a source term has been determined, onsite and offsite consequences are calculated for the atmospheric transport pathway. These consequences are then compared to evaluation guidelines for onsite consequences or release limits for offsite consequences for the identification of safety-class SSCs and TSRs.

#### 3.4.1.1 Source Term Composition (Radiological and Toxicological)

The bounding source term considered for the accident analyses is based on data for the fuel in the K East and K West Basins given in HNF-SD-SNF-SARR-005, HNF-SD-SNF-TI-009, 105-K Basin Material Design Basis Feed Description for Spent Nuclear Fuel Project Facilities defines an inventory for safety
analysis by considering inventories of Mark IV, Mark IA, and single-pass reactor fuel in the K Basins. High-burnup Mark IV fuel, the fuel type that results in the highest estimated dose to people exposed to the material, was selected as the bounding inventory for radiological dose calculations. Nuclear accountability records give the basis for the quantity, exposure variation and decay time variation of the stored fuel. The radionuclide inventory was estimated from these data.

The MCO contains finely divided particulate material associated with oxidation of the fuel. This material includes an oxide layer on the fuel and particulate remaining on fuel surfaces and in crevices after fuel washing and racking into the MCO as well as expected increases in oxidation products that occur during queuing at the K Basins and processing at the CVDF. The particulate inventory of the MCO dominates the airborne release. The radionuclide inventory of the sludge also is bounded by the high-burnup Mark IV fuel. The radionuclide content of sludge, based on sample analyses, is reported in Volume 2 of HNF-SD-SNF-TI-009. Comparison of the observed activity of sludge samples (activity per mass of uranium) with high burnup Mark IA fuel indicates that Mark IA fuel bounds the sludge observations for sludge samples.

Because any environmental release of SNF could have toxicological as well as radiological effects, both are computed for comparison with risk evaluation guidelines. From this comparison, the predominant risk of the spent fuel particles can be determined, and controls can be identified that prevent or mitigate both risks, thus simplifying the analysis and presentation. A detailed comparison of the toxicological and radiological hazards presented by the spent fuel particles has been performed (HNF-SD-SNF-TI-059). The basic assumptions used to show that the radiological risk guidelines are more limiting than the toxicological risk guidelines are listed below:

- The risk evaluation guidelines for toxicological and radiological hazards (Sellers 1997) are the foundation for determining the severity of a postulated airborne release under accident conditions. Both sets of guidelines cover onsite and offsite receptors and distinguish three accident frequency categories. The projected radiological consequences of an accident must be less than the appropriate radiological guideline. Similarly, the toxicological consequences of an accident must be less than the corresponding toxicological guideline.

- The primary material released under accident conditions is SNF particulate matter, which is mostly oxides of uranium. The safety basis composition of SNF (HNF-SD-SNF-TI-015) was used in the comparison. It is assumed that the bounding case accidents do not introduce toxic chemicals in addition to the particulate or change the relative toxicological versus radiological hazard of the particulate inventory used in the comparison (i.e., the particulate is assumed to be a single material with both radiological and toxicological effects). Note that chemical forms were assumed that would be most limiting. For added conservatism, the radiological dose factors were the largest allowed, and the air concentration limits were the smallest allowed. It has been shown that the radiological guidelines will be exceeded for a smaller mass of particulate released than the toxicological guidelines.
Because SNF contains no corrosive chemicals, a conservative exposure averaging time of 15 minutes was used in the calculation of average air concentration. Note that the time-weighted permissible exposure limit is based on an 8-hour averaging time while the emergency response planning guidelines are based on a 1-hour averaging time.

Air transport for very short durations is normally computed using a puff model. For the distances from release locations to the Site boundary, release durations less than 8 minutes could be modeled this way. Since the air concentration averaging time is 15 minutes, the puff model is not appropriate. Air transport is represented with a plume model for all release durations.

Long exposure times are a concern for chemical hazards. The air concentration guidelines use values with defined exposure times. If these times are exceeded, there is a potential for increased risk to the individuals downwind. The air concentration guidelines were conservatively reduced by the ratio of the assumed averaging time (15 minutes) to the release duration to account for this possibility.

3.4.1.2 Consequence Analysis. Radiological inhalation dose consequences for each analyzed accident were based on the following factors:

- Mass of material available for release
- Airborne release fraction (or airborne release rate and release time) and respirable fraction
- Leak path factor
- Atmospheric transport and dispersion of airborne particles
- Duration of exposure
- Breathing rates
- Dose conversion factors

The radiological dose to a maximum receptor of interest is typically determined by using the following equation:

\[ D = M \times \frac{Y}{Q'} \times BR \times UD \]
where

\[ D = \text{effective dose equivalent (rem)} \]
\[ M = \text{mass of respirable airborne material released (g)} \]
\[ \chi/Q' = \text{time-integrated atmospheric transport factor (s/m}^3) \]
\[ BR = \text{breathing rate (m}^3/\text{s})^1 \]
\[ UD = \text{dose per unit mass of radioactive material inhaled (rem/g)}^2 \]

The mass of respirable material released \((M)\) is determined by the specific accident scenario. The quantity is a function of the total airborne release fraction (or airborne release rate times release time) the respirable fraction and the leak path factor of any passive structural enclosure that may cause deposition of an airborne release before the release enters the atmosphere. The leak path factor is based on a time-integrated calculation of aerosol deposition within and release from an enclosure of given dimensions with specified leakage area pressure and temperature differentials. The mass of respirable radioactive particulate released \((M)\) during this is calculated using the following equation

\[ M = (\text{MAR})(\text{DR})(\text{RF})(\text{LPF}) \]

where

\[ \text{MAR} = \text{material at risk (kg)} \]
\[ \text{DR} = \text{damage ratio (conservatively assumed to be 1)} \]
\[ \text{RF} = \text{release fraction} \]
\[ \text{LPF} = \text{leak path factor (conservatively set to 1)} \]

The specific value of each parameter is determined in the individual DBA analyses and based on the physical phenomena of the accident. The composition of the material released \((M)\) is assumed to be the same as that of the K Basins fuel used to determine the unit dose \((UD)\). The accident conditions, as well as processing activities, are assumed to have minimal effect on the relative amounts of the various nuclides (HNF-SD-SNF-TI-059).

The atmospheric transport factor \((\chi/Q')\) is based on specific release conditions (e.g., ground level or elevated long or short duration) and the receptor's distance from the release. The atmospheric transport factor is the time-integrated normalized air concentration at the receptor's location. The transport factor includes the dilution of an airborne contaminant caused by atmospheric mixing and turbulence. The air transport values used have been generated using the GHQ computer program (WHC-SD-GN-SWD-30002 WHC-SD-GN-SWD-30003).

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\(^1\)The 12-hour breathing rate is \(3.33 \times 10^{-4}\) m²/s and the 24-hour breathing rate is \(2.64 \times 10^{-4}\) m²/s (HNF-SD-SNF-TI-059).

\(^2\)4.38 \times 10^5\) rem/g uranium (HNF-SD-SNF-TI-059).
Specific air transport values used to determine onsite and offsite consequences are provided in the facility FSAR Annexes.

Air transport factors are calculated using methods found in NRC Regulatory Guide 1 145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*. In each wind direction, the observed frequencies of particular wind speed and stability class combinations are used to compute a value that is exceeded only 0.5% of the time. This is repeated for all 16 compass directions to determine the worst-case location. In addition, the overall site value (exceeded only 5.0% of the time) was computed. For SNF Project facilities, the sector maximum (0.5%) is always greater than the overall site value (5.0%) for ground-level releases (HNF-SD-SNF-TI-009).

Exposures to the collocated worker onsite are bounded by the individual at the 100 m location. The risk evaluation guidelines apply to this individual. Exposures to members of the public are bounded by the individuals located on Washington State Highway 240 and at the Hanford Site boundary. For assessment purposes, DOE has directed (Sellers 1996) that the Hanford Site boundary be considered the location of the offsite receptor. Consequences at the Highway 240 location are included for reference only. Note that distances to the Hanford Site boundary and to the nearest public access locations onsite were computed using the methodology described in NRC Regulatory Guide 1 145. In this approach, the shortest distance in a 45° sector centered on the direction of interest is chosen for each of the 16 compass directions. Table 3-3 shows the maximum individual locations and air transport factors for ground-level releases.

The consequence dose analyses used in these accidents conservatively ignore adjustment of the air transport factors for the finite size of the source (i.e., building wake effects) or for the elevation of the release above ground level (i.e., stack effects). The outlet stack heights are less than twice the height of the building, which is an insufficient height to take credit for these effects. However, the duration of the release does affect the air transport values. Durations less than 1 hour use a point source model. Durations from 1 hour to 2 hours add the effect of plume meander. Air transport factors for longer periods (i.e., 12 and 24 hours) are calculated from the 2-hour and annual average values according to the method in NRC Regulatory Guide 1 145.

For accident analyses without controls, dose calculations for the maximum onsite individual assume that the individual remains at a distance of 100 m (328 ft) for the duration of plume passage up to a maximum of 12 hours. The 12-hour maximum duration is chosen because it is the normal shift for operating personnel. Dose calculations for the maximum offsite individual assume that the individual remains at the worst-case distance for the duration of plume passage up to a maximum of 24 hours. The 24-hour maximum duration is judged to be an appropriate endpoint for consequence calculations based on the premise that the offsite individual can be notified and appropriate corrective action taken within 24 hours of the start of the accident. For accident analyses with controls, a different exposure duration may be used based on a demonstrated ability to detect the accident and protect the receptor.
Table 3-3 Maximum Individual Locations and Air Transport Factors for Ground-Level Releases

<table>
<thead>
<tr>
<th>Receptor type</th>
<th>Air transport factors, s/m³</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>&lt;1 hr</td>
</tr>
<tr>
<td>K West Reactor and adjacent buildings&lt;sup&gt;b&lt;/sup&gt;</td>
<td></td>
</tr>
<tr>
<td>Onsite worker 100 m E</td>
<td>7.32 E-02</td>
</tr>
<tr>
<td>Columbia River 520 m W</td>
<td>3.55 E-03</td>
</tr>
<tr>
<td>100 Area Fire Station 3,750 m ESE</td>
<td>1.60 E-04</td>
</tr>
<tr>
<td>Hanford Site boundary 10,070 m W</td>
<td>4.49 E-05</td>
</tr>
<tr>
<td>Cold Vacuum Drying Facility</td>
<td></td>
</tr>
<tr>
<td>Onsite worker 100 m E</td>
<td>7.32 E-02</td>
</tr>
<tr>
<td>Columbia River 650 m W</td>
<td>2.44 E-03</td>
</tr>
<tr>
<td>100 Area Fire Station 3,750 m ESE</td>
<td>1.60 E-04</td>
</tr>
<tr>
<td>Hanford Site boundary 10,090 m W</td>
<td>4.48 E-05</td>
</tr>
<tr>
<td>Canister Storage Building</td>
<td></td>
</tr>
<tr>
<td>Onsite worker 100 m E</td>
<td>3.41 E-02</td>
</tr>
<tr>
<td>Highway 240 9,280 m W</td>
<td>2.36 E-05</td>
</tr>
<tr>
<td>Hanford Site boundary 17,390 m E</td>
<td>1.30 E-05</td>
</tr>
</tbody>
</table>

The 12 and 24 hour values are computed by interpolation. The rest are computed by GXQ (WHC SD GN SWD 30002 1995 GXQ Program Users Guide Rev 1 Westinghouse Hanford Company Richland Washington) All GXQ runs use a release height and receptor height of zero meters.<br><sup>a</sup>Use of K West represents bounding values for both K Basins.<br><sup>b</sup>For the Canister Storage Building  this receptor is located 100 m ESE.<br><sup>c</sup>For the Canister Storage Building  this receptor is located 12 670 m E.<br><sup>d</sup>For the Canister Storage Building  this receptor is located 17 390 m ESE.<br><sup>e</sup>NA = not applicable.
Inhalation rates (BR) for the reference man (ICRP Publication 23) are used. The light activity breathing rate $(3.33 \times 10^{-4} \text{ m}^3/\text{s})$ normally applies during the first 16 hours of the day when the person is assumed to be awake. The resting breathing rate $(1.24 \times 10^{-4} \text{ m}^3/\text{s})$ applies during the last 8 hours of the day when the person is assumed to be asleep. The 24-hour average breathing rate is $2.64 \times 10^{-4} \text{ m}^3/\text{s}$.

The dose per unit mass of radioactive material is the 50-year dose commitment for all relevant exposure pathways per gram of radioactive material inhaled. The major radiation exposure pathway for the identified accidents is inhalation of radioactive material. Dose contributions from the submersion pathway were calculated and found to be negligible with respect to the total dose for the radionuclides of interest (Hey 1995). Doses from groundshine also are expected to be negligible because most of the radionuclides of interest are alpha emitters. Therefore, the doses from groundshine and submersion are not included in the radiological dose calculations.

Potential doses from the ingestion pathway are not considered because DOE, state, and federal emergency preparedness plans limit ingestion of contaminated food in the event of an accident. DOE/RL-94-02, *Hanford Emergency Response Plan*, governs emergency response for all Hanford Site facilities. The primary determinant of exposure from the ingestion pathway is the effectiveness of public health measures (i.e., interdiction) rather than the severity of the accident itself. Ingestion, if it occurs, involves a relatively slow-to-develop pathway and is not considered an immediate threat to an exposed population in the same sense as the inhalation pathway. In addition, calculations in HNF-SD-SNF-TI-059, *A Discussion on the Methodology for Calculating Radiological and Toxicological Consequences for the Spent Nuclear Fuel Project at the Hanford Site*, show that the contribution of ingestion to the total dose is negligible compared to the inhalation contribution.

The composition of the K Basins fuel decayed to May 31, 1998 was used to determine the committed effective dose equivalent per gram $(4.38 \times 10^5 \text{ rem/g})$ of respirable release (HNF-SD-SNF-TI-059). Isotopes of plutonium, americium, and curium constitute 99.5% of the total inhalation dose. The values for the dose per unit respirable radioactive material inhaled (UD) are the product of the activities and the dose conversion factor found in EPA Federal Guidance Report Number 11 *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation Submersion and Ingestion*. The dose conversion factor for tritium was increased by 50% to account for skin absorption (ICRP Publication 30). The committed effective dose equivalent for $^{40}$Kr is the submersion dose factor from EPA Federal Guidance Report Number 12 *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*. The value of the dose conversion factor divided by the light activity breathing rate.

**3.4.3 Toxicological Effects** The SNF is primarily uranium metal, which is known to have toxicological effects. Plutonium and other transuranic heavy metals also are present in small quantities but add little to the overall toxicity of the fuel. For example, if the toxic air concentration limits for uranium are applied to neptunium, plutonium, americium, and curium, the sum-of-fractions' indicator of toxicity increases about 0.2% (HNF-SD-SNF-TI-059). Thus, the uranium content of SNF controls the potential health impacts downwind following a postulated accident.
accident. Uranium acts like many heavy metals to damage one or more internal organs of individuals exposed to high air concentrations. The toxicity depends on the solubility of the uranium, with more soluble compounds being a greater hazard because they are transferred from the respiratory tract into the blood more quickly. The chemical form, of radiological isotopes that is most soluble (toxic) was assumed when determining the toxicological effects.

No routine chemical processes are conducted in the SNF Project facilities. Purging and backfilling the MCOs involves the use of an inert gas (helium). Some chemicals such as those used for equipment decontamination, may be used occasionally (HNF-SD-SNF-CM-001). However, there are no chemical inventories of concern for safety analysis considerations.

3.4.1.4 Frequency Estimates. The frequency and consequence of each design basis or evaluation basis accident is to be evaluated to determine the effectiveness of selected preventive and mitigative controls. Thus each DBA is evaluated and a frequency assessment made to determine the frequency of the initiating event and subsequent probabilities of failures required to result in the identified consequence. This evaluation of the frequency of the total accident scenario is documented for each DBA represented in the accident analysis. The frequency and consequence for each DBA are then used to determine whether the identified controls are sufficient to meet DOE guidelines. This process develops quantitative frequencies for the DBAs and results in qualitative probabilities for the remaining accidents identified in the hazard analysis.

3.4.1.5 Risk Guidelines. The DOE-recommended radiological risk evaluation guidelines (Sellers 1997) are shown in Table 3-4. These risk evaluation guidelines are used for identifying safety-class and safety-significant SSCs, they implement the guidance of DOE Order 6430.1A, Section 1300.14 ‘Guidance on Limiting Exposure of the Public,’ and are consistent with the graded approach to safety required by DOE Order 5480.23.

<table>
<thead>
<tr>
<th>Event category</th>
<th>Frequency range (per year)</th>
<th>Onsite risk evaluation guidelines* (rem)</th>
<th>Offsite accident release limits* (rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Anticipated</td>
<td>1.0 E-01 to 1.0 E-02</td>
<td>1</td>
<td>0.5</td>
</tr>
<tr>
<td>Unlikely</td>
<td>1.0 E-02 to 1.0 E-04</td>
<td>10</td>
<td>5.0</td>
</tr>
<tr>
<td>Extremely unlikely</td>
<td>1.0 E-04 to 1.0 E-06</td>
<td>25</td>
<td>5.0</td>
</tr>
</tbody>
</table>

Note: All doses are committed effective dose equivalents.

According to HNF-PRO-704, *Hazard and Accident Analysis Process*, satisfaction of the radiological evaluation guidelines in HNF-PRO-704 meets the goals of SEN-35-91 *Nuclear Safety Policy*. However, the guidelines used by the SNF Project (Sellers 1997) are more conservative and more restrictive and bound by the guidelines specified in HNF-PRO-704, Table D-1. As such, satisfaction of the SNF Project radiological evaluation guidelines (Sellers 1997) will meet the goals of SEN-35-91. As discussed in future sections, the mitigated DBAs using the identified preventive and mitigative SSC features met the SNF Project radiological evaluation guidelines (Sellers 1997) and, consequently, the goals of SEN-35-91.

3.4.1.6 Safety Structures, Systems, and Components

"Safety class" and "safety significant," as related to safety SSCs, are defined consistently for the SNF Project. Safety-class SSCs prevent or mitigate releases to the public that would otherwise exceed the offsite radiological evaluation limits or they prevent accidental nuclear criticality. Specifically, DOE Order 6430.1A Section 1300-3.2 states, "Safety-class items are systems components and structures, including portions of process systems whose failure could adversely affect the environment or safety and health of the public. Specifically, safety-class items are those systems components and structures with the following characteristics:

- Those whose failure would produce exposure consequences that would exceed the guidelines in Section 1300-1.4, Guidance on Limiting Exposure of the Public at the site boundary or nearest point of public access
- Those required to maintain operating parameters within the safety limits specified in the operational safety requirements during normal operations and anticipated operational occurrences
- Those required for nuclear criticality safety
- Those required to monitor the release of radioactive materials to the environment during and after a DBA
- Those required to achieve and maintain the facility in a safe shutdown condition
- Those that control the safety-class items described above [DOE Order 6430.1A, Section 1300-3.2, Safety Class Items]

A safety-class SSC satisfies the criteria of DOE Order 6430.1A (Sellers 1997) if the SSC does one of the following:

1. Prevents or mitigates offsite doses in excess of 500 mrem (5 mSv) total effective dose equivalent
2. Places or maintains an operating process in a safe condition that prevents or mitigates offsite doses in excess of 500 mrem (5 mSv) total effective dose equivalent
3 Monitors the release of radioactive materials to the environment during and after accidents in which the monitor's output initiates emergency response plan actions or operator actions to place the operating process in a safe condition in accordance with criterion 2

4 Maintains criticality controls

5 Supports the safety function of a safety-class SSC, this includes control and monitoring functions (e.g. operating air, electrical power, instrumentation)

Safety-significant SSCs prevent or mitigate releases of radiological materials or toxic chemicals to onsite workers. This includes barriers that are judged to substantially contribute to defense in depth independent of quantitative analysis. Safety significant also describes worker safety SSCs that protect the facility worker from serious injury caused by hazards not controlled by institutional safety programs. Specifically HNF-PRO-704 implements criteria for safety-significant SSCs as follows:

6 Prevents or mitigates a radiological dose or chemical exposure that challenges the risk evaluation guidelines in Letter 97-SFD-172 (Sellers 1997)

7 Places or maintains an operating process in a safe condition that prevents or mitigates consequences that exceed criterion 6 listed above

8 Prevents or mitigates exposure in excess of 5 rem (50 mSv) total effective dose equivalent or an airborne concentration in excess of Emergency Response Planning Guidelines-2 limit to facility operators who are relied on to achieve the safe condition of criterion 2 or criterion 7 listed above

9 Monitors the release of radioactive and/or hazardous materials to the environment during and after accidents where the monitor's output initiates emergency response plan actions or operator actions to place the operating process in a safe condition in accordance with criterion 7

10 Supports the safety function of a safety-significant SSC this includes control and monitoring functions (e.g. operating air, electrical power, instrumentation)

11 Prevents or mitigates an acute fatality to a facility worker or serious injury to a group of workers, except where the SSCs are controlled through an implemented institutional safety or radiation protection program

12 Provides defense-in-depth prevention or mitigation of an uncontrolled release of radioactive and/or hazardous material deemed significant in the safety analysis

For each accident scenario, the airborne radiological dose calculated using the methods described here is compared with the appropriate onsite and offsite evaluation guidelines and
limits If the radiological dose for the unmitigated case exceeds the guideline, mitigating safety features with appropriate safety-class and/or safety-significant functional classification, are described. The dose consequences are recalculated taking appropriate credit for the mitigating safety features to verify that the mitigated doses satisfy the guidelines.

3.4.1.7 Multi-Canister Overpack Particulate Inventory and Other Common Design Basis Accident Assumptions

The safety basis MCO particulate inventory used in the accident analyses and other assumptions has some relevance to all the DBAs and is listed here to establish consistency between individual accident analysis chapters.

- The safety basis loose particulate inventory is 0 kg immediately after washing at the K Basins.
- The safety basis particulate inventory for an MCO is 15 kg at the end of draining at the CVDF and up to 25 kg after processing.
- The safety basis particulate inventory for an MCO at the CSB after 40 years of storage is 34 kg (HNF-SD-SNF-TI-015).

3.4.2 Design Basis Accidents

According to DOE Order 5480.23, DBAs are those accidents that are postulated for the purpose of establishing functional requirements for safety-class and safety-significant SSCs. The DBAs have been analyzed to quantify consequences and compare them with evaluation guidelines. The process is iterative, starting by taking no credit for mitigative features and comparing the results to the evaluation guidelines. Credit is then taken for safety SSCs that prevent or mitigate the consequences to show that the results are below the evaluation guidelines. The process continues after the evaluation guidelines are met by identifying other SSCs that, while not designated as safety class, provide additional mitigative features as defense in depth. Finally, identified safety-class or safety-significant SSCs must be defined to have safety functions to meet the applicable functional preventive or mitigative requirements (if any) to address the consequences of the DBA. The DBAs, the quantified consequences, and the comparisons with evaluation guidelines are provided in the facility FSAR Annexes. In addition, each individual DBA section references a supporting calculational note that provides more detail.

3.4.3 Beyond Design Basis Accidents

DOE Order 5480.23 Attachment 1, paragraph 4 f (3)(d) currently requires consideration of beyond DBAs for nuclear facilities but presents no clear guidelines as to how or to what criteria they should be evaluated and judged. DOE-STD-3009-94 indicates that the purpose of beyond DBA presentation is to gain insight into the magnitude of these events particularly if they are close in frequency to pertinent DBAs but beyond the credible range and have consequences that exceed evaluation guidelines. While these events would be beyond...
requirements of further safety-class or safety-significant functions, they might provide guidance on the prioritization of long-term safety improvements for a facility. The Standard specifically excludes evaluation of human-generated external events as beyond DBAs. Beyond DBAs address issues such as the following:

- Breach of a low-heat MCO during receiving caused by excessively cold weather and resulting ice formation
- Collapse of the building superstructure as a result of natural phenomena hazards of a frequency lower than the facility's current natural phenomena hazard design basis events
- DBAs currently included in Sections 3 4 2 of the facility FSAR Annexes with the assumption that safety-class SSCs currently credited have failed. Examining the unmitigated consequences of these events as presented in the facility FSAR Annexes provides insight into the magnitude of the consequences of the events.

The results from analyses of beyond DBAs are provided in the facility FSAR Annexes.

3.5 REFERENCES


DOE Order 5480 22 Technical Safety Requirements U S Department of Energy, Washington D C

DOE Order 5480 23 Nuclear Safety Analysis Reports U S Department of Energy, Washington D C

DOE Order 6430 1A, General Design Criteria U S Department of Energy, Washington D C


Figure 3-1 Three-by-Three Likelihood and Consequence Ranking Matrix

Combinations that identify situations of major concern

Combinations that identify situations of concern

Note: This consequence ranking matrix is based on Figure 3-3 in DOE-STD-3009-94
Figure 3-2 Methodology for Identification and Analysis of Abnormal Events
CHAPTER 40

SAFETY STRUCTURES, SYSTEMS, AND COMPONENTS
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LIST OF TERMS

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<th>Acronym</th>
<th>Full Form</th>
</tr>
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<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
</tr>
<tr>
<td>FSAR</td>
<td>Final Safety Analysis Report</td>
</tr>
<tr>
<td>NRC</td>
<td>U.S. Nuclear Regulatory Commission</td>
</tr>
<tr>
<td>SNF</td>
<td>Spent Nuclear Fuel</td>
</tr>
<tr>
<td>SSC</td>
<td>Structure System and Component</td>
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4 0 SAFETY STRUCTURES, SYSTEMS, AND COMPONENTS

4 1 INTRODUCTION

The purpose of this chapter is to provide a general overview description of the methodology and criteria used to identify and select Spent Nuclear Fuel (SNF) Project structures, systems, and components (SSCs) that are necessary to ensure public and onsite worker safety, provide defense-in-depth, or contribute to facility worker safety or institutional control as identified in Chapter 3 0. Additional information specific to individual SNF Project facilities is provided in the facility annexes to the SNF Project Final Safety Analysis Report (FSAR). This chapter is organized and prepared in accordance with the requirements of DOE Order 5480 23, Nuclear Safety Analysis Reports, as amplified by DOE-STD-3009-94, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports.

The SNF Project facilities are designated hazard category 2 nuclear facilities with the potential for an accident resulting in significant onsite and offsite consequences. Hazard category 2 facilities are designed with safety-significant SSCs to provide protection to the onsite workers, and as appropriate, safety-class SSCs to provide protection to the offsite public. The public protection function requires that safety-class SSCs receive more formality to establish functional requirements and performance criteria than safety-significant SSCs.

Each accident analysis section in Chapters 3 0 of the facility FSAR Annexes concludes with a summary of safety SSCs that provides the basis for FSAR Annex Chapters 4 0. A summary listing of the accident categories and the designated safety SSCs that prevent or mitigate their consequences is provided in the facility FSAR Annexes. Also provided are a summary list of the safety SSCs identified and the level of safety provided for each accident category. Many SSCs provide a safety function for prevention or mitigation of more than one accident. Detailed definitions of safety-class and safety-significant SSCs are provided in Section 3 4 1 6.

4 2 REQUIREMENTS

The requirements that form the basis for selection of safety SSCs are found in HNF-SD-SNF-RD-001, Spent Nuclear Fuel Project Standards/Requirements Identification Document. The specific requirements applicable to this chapter include:

- DOE Order 6430 1A, General Design Criteria
- DOE Order 5480 23, Nuclear Safety Analysis Reports
- DOE Order 5480 28, Natural Phenomena Hazards Mitigation
- DOE Letter 97-SFD-172, Risk Evaluation Guidelines (REGs) to Ensure Inherently Safer Designs (Sellers 1997)
In Letter 95-SFD-167 Implementation of the K Basins Spent Nuclear Fuel Project (SNFP) Regulatory Policy (Sellers 1995) the U.S. Department of Energy (DOE) establishes the requirement that new SNF Project facilities such as the Cold Vacuum Drying Facility and Canister Storage Building achieve "nuclear safety equivalency" to comparable U.S. Nuclear Regulatory Commission (NRC)-licensed facilities. The SNF Project identified the NRC requirements that were needed in addition to existing and applicable DOE requirements to establish nuclear safety equivalency. These NRC requirements and the process used to identify them are documented in HNF-SD-SNF-DB-003 spent Nuclear Fuel Project Path Forward Additional NRC Requirements WHC-SD-SNF-DB-010, Cold Vacuum Drying System Natural Phenomena Hazards and in NRC Regulatory Guide 3.48, Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage). These documents establish specific design requirements for the Cold Vacuum Drying Facility and Canister Storage Building.

4.3 SAFETY-CLASS STRUCTURES, SYSTEMS, AND COMPONENTS

The following paragraphs describe the processes used to identify safety-class SSCs for the SNF Project facilities. Refer to the facility FSAR Annexes for the specific safety-class SSCs. Hazard category 2 facilities do not normally have the consequence potential associated with high hazard facilities such as Class A reactors. As described in DOE-STD-3009-94 for the SNF Project facilities the same level of information is not expected to be provided in the FSAR as is expected for Class A reactors. The facility FSAR Annexes include determination of safety functions and functional requirements for safety SSCs and designation of performance criteria. The FSAR is focused on identifying functional requirements that, in general, are neither absolute nor subject to fine safety margin resolution. Associated performance criteria are only defined for critical operational aspects of SSCs and not general design requirements. Safety analyses and design information for the safety-class SSCs are provided in each facility FSAR Annex. In accordance with DOE Order 6430.1A, safety-class SSCs are subject to appropriately higher quality design, fabrication, and test standards and codes to assure reliability and allow credit for capabilities assumed in the safety analysis. Safety-class SSCs are designed to the Boiler and Pressure Vessel Code Section III Class II (ASME 1995), or to other comparable safety-related codes and standards appropriate for the system being designed. See the facility FSAR Annexes for specific applicable codes and standards.

DOE Letter 97-SFD-172 (Sellers 1997) states that identification of safety-class engineered safety features for the SNF Project shall remain consistent with DOE Order 6430.1A. This Order defines safety-class SSCs including portions of process systems as those SSCs whose failure could adversely affect the environment or the health and safety of the public. Detailed definitions of safety-class SSCs are provided in Section 34.16.

The SSCs credited with a safety-class function in Chapters 30 of the facility FSAR Annexes are detailed in Sections 4.3 of the facility FSAR Annexes. The sections also include details regarding the representative and bounding accident for each safety-class SSC, safety
function codes and standards, system description, functional requirements system evaluation, and controls (technical safety requirements)

DOE Order 6430 1A identifies aspects related to safety-class SSCs that must be addressed to a higher quality standard. These are described in the following paragraphs.

**Single-Failure Criterion and Redundancy**

The design shall ensure that a single failure (as defined in DOE Order 6430 1A) does not result in the loss of capability of a safety-class system to accomplish its required safety functions. To protect against single failures, the design shall include appropriate redundancy and shall consider diversity to minimize the possibility of concurrent common-mode failures of redundant items.

**Equipment Environment Considerations**

- **General**

  Safety-class items shall be designed to withstand the effects of, and be compatible with, the environmental conditions associated with operation, maintenance, shutdown testing, and accidents. The environmental capability of equipment shall be demonstrated by appropriate testing, analysis, and operating experience, or other methods that can be supported by auditable documentation or a combination of these methods.

- **Environmental Qualification of Equipment**

  Equipment qualification shall provide assurance that safety-class items will be capable of performing required safety functions under design basis accident conditions. The qualification shall demonstrate that the equipment can at a minimum perform for the period of time that its safety functions are required. Environmental temperature, pressure, and humidity shall be based on the most severe postulated accident affecting the particular item.

- **Equipment Operability Qualification**

  Testing or a combination of testing and analysis shall be the preferred method of demonstrating the operability of fluid system components, mechanical equipment, instrumentation, and electrical equipment that are required to operate during and following a design basis earthquake.
Maintenance

The design shall consider the maintainability factors peculiar to the specific equipment to be used in the facility. Facility design shall provide for routine maintenance, repair, or replacement of equipment subject to failure.

Safety-class items shall be designed to allow inspection, maintenance, and testing to ensure their continued functioning, readiness for operation, and accuracy.

The design of all process equipment shall include features to minimize self-contamination of the equipment, piping, and confinement areas. The design of process equipment shall also include features to minimize the spread of contamination out of local areas.

Testing

The design shall include provisions for periodic testing of monitoring, surveillance, and alarm systems. In addition, the design shall provide the capability to periodically test under simulated emergency conditions safety-class items required to function under emergency conditions.

4.4 SAFETY-SIGNIFICANT STRUCTURES, SYSTEMS, AND COMPONENTS

DOE-STD-3009-94 states that in general, safety-class SSCs require more formality in establishing functional requirements and performance criteria than safety-significant SSCs because of their public protection function. For hazard category 2 facilities, there is potential for an accident resulting in significant onsite consequences which may have consequences offsite. These facilities characteristically have safety-significant SSCs. SSCs designated as safety significant are designed to meet the single-failure criteria of safety-class SSCs and secondary confinement where the potential exists for significant onsite consequences which may have offsite consequences. As with safety-class SSCs, safety-significant performance criteria are only defined for critical aspects of SSCs not general design.

DOE Letter 97-SFD-172 (Sellers 1997) provides a rigid definition of safety-significant SSCs. This letter states that NRC Regulatory Guide 3.48 requires an evaluation of accidents to workers (defined in the letter to be collocated worker within the controlled area). Location of the collocated worker varies at the Hanford Site. Consequently, the determination is made that the collocated worker be 100 m from the facility or that distance which results in the maximum calculated exposure to the collocated worker thereby allowing minimization of actual exposures. This location will be used for the designation of safety-significant SSCs to protect collocated workers and to provide defense in depth. DOE Letter 97-SFD-172 (Sellers 1997) further states that identification of safety-significant engineered safety features shall remain consistent with DOE-STD-3009-94 but with the additional conservatism to address the collocated worker.
Finally, the letter states that onsite risk evaluation guidelines will be employed to identify safety-significant SSCs to protect the collocated workers.

DOE Order 6430 1A identifies several aspects related to safety-class SSCs that must be addressed to a higher quality standard (see Section 4 3). These aspects related to design, qualification, operability, maintenance, and testing are applied to the extent possible to safety-significant SSCs.

The SSCs credited with a safety-significant function in Chapters 3 0 of the facility FSAR Annexes are detailed in Sections 4 4 of the facility FSAR Annexes or are combined in annex Sections 4 3 and 4 4 as appropriate for ease of presentation.

4.5 REFERENCES


DOE Order 5480 28, *Natural Phenomena Hazards Mitigation* U S Department of Energy, Washington, D C

DOE Order 6430 1A, *General Design Criteria* U S Department of Energy Washington D C


CHAPTER 5 0

DERIVATION OF TECHNICAL SAFETY REQUIREMENTS
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November 1999
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<td>AC</td>
<td>Administrative Control</td>
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<td>CSB</td>
<td>Canister Storage Building</td>
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<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
</tr>
<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
</tr>
<tr>
<td>DOE-RL</td>
<td>U.S. Department of Energy, Richland Operations Office</td>
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<tr>
<td>FSAR</td>
<td>Final Safety Analysis Report</td>
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<tr>
<td>LCO</td>
<td>Limiting Condition for Operation</td>
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<td>LCS</td>
<td>Limiting Control Setting</td>
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<td>MCO</td>
<td>Multi-canister overpack</td>
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<td>SNF</td>
<td>Spent Nuclear Fuel</td>
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<tr>
<td>SSC</td>
<td>Structure, System and Component</td>
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<td>TSR</td>
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5.0 DERIVATION OF TECHNICAL SAFETY REQUIREMENTS

5.1 INTRODUCTION

This Chapter consists of summaries and references to pertinent sections of the safety analysis report that are used to describe design and administrative features needed to prevent or reduce the consequences of an accident. The Safety Limits, Limiting Control Settings (LCSs), Limiting Conditions for Operation (LCOs), Surveillance Requirements, Administrative Controls (ACs) and Design Features form the basis of the Technical Safety Requirement (TSR) document and provide the logical link between the TSRs and the safety analysis report.

As discussed in Chapter 3.0 of the facility annexes to the Spent Nuclear Fuel (SNF) Project Final Safety Analysis Report (FSAR), a hazard categorization process assessed the hazardous material inventory at risk for release, unmitigated by any safety features. The TSRs for each of the facilities were developed based on the graded approach applied to the hazard and accident analyses and the final hazard category 2 designation for SNF Project facilities.

This Chapter includes the following information:

- Qualitative and quantitative TSR selection criteria
- Derivation logic for minimum staffing levels
- Identification of TSR interfaces with other Hanford Site facilities

The facility FSAR Annexes include the following facility-specific information:

- Operational modes that designate distinguishable facility configurations and operational conditions
- Derivation discussion of the TSR controls selected
- A table that links the Chapter 3.0 accident analyses and the TSRs
- A list of Design Features not covered by the TSRs

The necessary and sufficient preventive and mitigative features determined to be essential in Chapter 3.0, "Hazard and Accident Analyses," and described in Chapter 4.0 "Safety Structures, Systems and Components," are identified in the facility FSAR Annex chapters. Information necessary for preparing the separate TSR document required by U.S. Department of Energy (DOE) Order 5480.22, "Technical Safety Requirements," also is provided in these chapters.

TSRs define acceptable conditions, safe boundaries, and management or ACs that ensure safe operation of a nuclear facility and reduce the potential risk to the public and onsite workers from uncontrolled releases of radioactive material or from radiation exposures caused by inadvertent criticality (DOE Order 5480.22). As established in Chapter 3.0 of the facility FSAR...
Annexes the toxicological hazards of the radionuclide inventory are bounded by the radiological consequences. Therefore, control of the radiological risk similarly controls any lesser risk from toxicological hazards.

5.2 REQUIREMENTS

The requirements that form the basis for developing TSRs are found in HNF-SD-SNFRD-001, *Spent Nuclear Fuel Project Standards/Requirements Identification Document*. Specific requirements applicable for this Chapter include the following:


5.3 TECHNICAL SAFETY REQUIREMENTS COVERAGE

The TSRs for analyzed hazards and accidents are summarized in Tables 5-1 of the facility FSAR Annexes. This table lists TSR controls in accordance with the accident analyses for each facility FSAR Annex. Chapter 3. Table 5-1 is written to provide a road map from the respective accident analysis section to the relevant subheadings within Section 5.5 of the facility FSAR Annexes, where derivation details are arranged by TSR control.

The necessary and sufficient TSR controls are established based upon consideration for public safety, significant defense in depth, significant worker safety, and for maintaining radiological consequences below risk guidelines.

5.3.1 Criteria

Chapter 3.0, Section 3.3.1 details the control selection process from initial hazard analysis identification to selection of hardware or administrative programs for controls. Guidance for selecting TSR candidates is found in Appendix A of HNF-PRO-700, *Safety Analysis and Technical Safety Requirements*. The criteria used for selecting TSR candidate controls are outlined in Section 5.3.11 and the criteria for determining the level of selected controls are outlined in Sections 5.3.1.2 and 5.3.1.3. Generalized criteria may be found in DOE orders. The following criteria, while considered to meet the generalized order criteria, reflect an interpretive base determined by coordination with other DOE site TSR writing personnel, a...
DOE-sponsored TSR training course, contractor procedures (HNF-PRO-700) reactor standard technical specifications (NUREG-1431), and site expertise

5 3 1 1 Control Selection Criteria  The following elements were considered in the selection of controls

- Analysis Assumptions — Controls were selected to protect major analysis assumptions if the assumption significantly affected consequences

- Incredible Accidents — Controls were selected if required to prevent an accident (i.e., caused the accident to be judged incredible) when an unmitigated consequence analysis was not performed

- Sensitivity Analysis — Controls were selected for less significant analysis assumptions if variability of the assumption significantly affected consequences

- Optimization — Controls were selected that could address multiple accidents to minimize implementation costs

- Priorities — Controls were selected first to prevent, then to mitigate first to involve passive systems, then to involve actively engineered systems, and first to use engineered controls then to use administrative programs

- Practical Safety-Class and Safety-Significant Designation — Controls were selected that would have value in being designated safety class or safety significant

- Risk Guidelines — Controls were identified that would bring the frequency or consequences within onsite risk evaluation guidelines and/or offsite accident release limits identified in Chapter 3.0 of the facility FSAR Annexes

- Human Factors — Controls were selected that took into account human factors considerations

- Facility Worker Safety — Controls were selected that would not significantly increase exposures to the facility worker or would significantly minimize serious risk to facility workers as discussed in Chapter 3.0

- Significant Defense-in-Depth — Controls were selected from the suite of defense-in-depth items if they significantly minimized risk as discussed in Chapter 3.0

- Operational Flexibility — Controls were selected that minimized the need for additional operators or maintenance personnel

5 3 1 2 Safety Limits and Limiting Control Settings  DOE-STD-3009-94 states that SLs [Safety Limits], if used, are reserved for a small set of extremely significant features that prevent
potentially major offsite impact. Hanford Site-specific procedures extend this SL applicability to those features that prevent potentially major onsite impact as well. Criteria for the selection of Safety Limits are established as the following set of elements which must be all true before a Safety Limit is selected for any of the defined bounding accidents:

- A primary passive barrier failed (e.g., multi-canister overpack [MCO])
- Passive barrier failure was the direct result of exceeding a physical parameter (e.g., system pressures sufficient to cause catastrophic failure of the MCO)
- This physical parameter can be directly measured by field personnel (e.g., tempered water temperatures)
- Radiological consequences without controls exceeds either onsite risk evaluation guidelines or offsite accident release limits
- The physical parameter limit prevents an accident from occurring rather than mitigating an accident after it has already occurred.

LCSs are setpoints on safety systems that control process variables to prevent exceeding Safety Limits. The specific setpoints are chosen such that if exceeded, sufficient time is available to automatically or manually correct the condition before exceeding Safety Limits. The LCSs are normally combined with their respective LCOs. By combining the LCSs with the LCOs, the LCS setpoint becomes part of the operability of the system. Furthermore, safety is enhanced by placing the applicability actions, and surveillance for a system in a single location, and the complexity of the TSR document is reduced.

5.3.1.3 Limiting Conditions for Operations and Administrative Controls Controls for accidents and hazards exceeding offsite or onsite risk guidelines which are not candidates for Safety Limits according to the criteria listed above are candidates for either LCOs or ACs. ACs are considered equally important to the control philosophy as LCOs. They are not treated as less meaningful controls for safety.

LCOs are prepared for systems, equipment, or conditions that provide safety functions and meet one or more of the following descriptions:

- Installed instrumentation that is used to detect and indicate a criticality accident or a significant degradation of physical barriers to the release of radiological material.
- Structures, systems, and components (SSCs) that are relied upon in the safety analyses to prevent or mitigate accidents or transients that involve the assumed failure or present a challenge to, the integrity of a physical barrier to the release of radiological or hazardous material.
- Process variables or environmental or facility conditions that are initial conditions for those design basis accidents or transient analyses that involve the assumed failure of or present a challenge to, the integrity of a physical barrier

- Experiments and experimental facilities that could provide a path through barriers to the release of radiological or hazardous material or that affect criticality safety

- Systems and equipment that are used for handling fissile material when identified in the accident analyses as being part of the primary success path to providing an acceptable risk of facility operations

In addition, ACs are established as necessary to support operating limits provided by Safety Limits, LCSs and LCOs and to provide requirements that maintain the safety basis of the facility as described in the safety basis documentation

The requirements contained in Chapters 6 through 17 that form the basis for the AC programs are contractual requirements. Many of the programs may be administered by contractor organizations that are outside of the SNF Project organizational structure. The minimum requirements for each AC program are found in the program key elements section of the TSR document. However, a noncompliance within a specific procedure that implements an AC program is not necessarily a TSR violation. A TSR violation occurs as a result of failure to comply with an AC program (not necessarily individual implementing procedures) or the intent of an AC program

5.4.4 Surveillance Requirements Each LCO identified in the facility FSAR Annexes (Section 5.5.X) includes the information necessary to derive the appropriate surveillance requirements

5.3.2 Safety Structures, Systems, and Components not Provided with Technical Safety Requirement Coverage

The safety-class and safety-significant SSCs that have not been provided with TSR coverage are identified in the facility FSAR Annexes. All other safety SSCs are provided with TSR coverage with an LCO or are included in an AC program. The bases for not providing TSR coverage are provided in the facility FSAR Annexes

5.4 DERIVATION OF FACILITY MODES

5.4.1 Operational Modes

The facility operational modes for the SNF Project facilities are identified to indicate overall facility status. See Sections 5.4.1 of the facility FSAR Annexes for the facility-specific mode definitions
5 4 2 Minimum Staffing Levels

The minimum operations shift complement in the facility modes is established in each facility FSAR Annex and is based upon the minimum staff in each mode considered adequate to perform the minimum safety functions necessary to protect the health and safety of the public onsite workers, and the environment during normal operations, and abnormal and emergency conditions. Consideration is given to the human factors principles addressed in Chapter 13 that guide day-to-day operations.

Determination of the minimum staff is based on ensuring that the following safety criteria are met:

- TSR compliance
- Emergency initial notification and initial response

The minimum staff does not include individuals necessary to fulfill the SNF Project mission goals and objectives, or to meet all other safety, environmental and authorization basis requirements and commitments.

Qualification training for the minimum staff (managers, engineers, operators, and health physics technicians) is addressed in Chapter 12. The program for TSR, emergency, and alarm response administrative procedures is also addressed in Chapter 12. Emergency response is addressed in Chapter 15. The bases for determining the minimum staff are provided in the following paragraphs:

Normal Operations — The minimum staff during normal operations is necessary to (1) safely operate the facility, (2) perform required TSR surveillance to ensure minimum TSR compliance (3) provide radiological control, and (4) perform maintenance of safety SSCs. Surveillance, such as system calibrations and functional tests, and AC program commitments are planned and scheduled to ensure TSR compliance.

Abnormal Conditions — The minimum staff during abnormal conditions is necessary to perform required actions specified in LCO action statements with completion times of "immediately" or less than 8 hours to ensure TSR compliance. Hanford Site experience has shown that additional staff could be provided within 8 hours if needed (considering the most adverse weather and travel conditions) to ensure all LCO completion times are met. LCO completion times of "immediately" imply the highest sense of urgency and are given top priority over all other activities. The minimum staff is not given tasks that could interfere with meeting TSR requirements.

Emergency Conditions — The minimum staff during emergency conditions is necessary to respond to the spectrum of accidents analyzed in Chapter 3 of the facility FSAR Annexes. The minimum staff must make prompt initial notifications and implement initial protective actions to preclude or reduce the exposure of individuals affected by hazards or unsafe conditions.
conditions during an emergency. Specific functions that must be performed by the minimum staff in an emergency include:

- Implementation of alarm response, plant response, and emergency management procedures
- Classification of events
- Initial prompt notifications
- Performance of administrative functions, such as preparing occurrence reports
- Communication of facility status and response to questions
- Support for the U.S. Department of Energy, Richland Operations Office (DOE-RL) Emergency Operations Center

5.5 TECHNICAL SAFETY REQUIREMENT DERIVATION

This section details level of control and derivation bases for the TSRs. This section is organized by control and includes a derivation discussion for those surveillance associated with each specific control. It can be used as an interface to link the accident analyses and the TSRs. It defines the reasons controls were chosen, including details about AC bases that are not discussed in the TSR document. Tables 5-1 of the facility FSAR Annexes provide a road map between the accidents in Chapters 30 and the TSR controls derived from them.

All controls are discussed within their applicable facility FSAR Annexes except for Sections AC 5.1 through 5.6 of the separate TSR documents. AC 5.1, "Purpose," AC 5.2, "Contractor Responsibility," and AC 5.3, "Compliance," define the need for subsequent AC provisions and clarify responsibility according to good management practice. AC 5.4, "Technical Safety Requirement Violation," AC 5.5, "Occurrence Reporting," and AC 5.6, "Organization," satisfy the requirements of DOE Order 5480.22 Section 9.3 (5). This DOE order section requires ACs for reporting deviations and identifying staffing requirements. Worker safety program requirements are addressed in other FSAR sections based on design and operations information.

5.6 DESIGN FEATURES

Design Features are those features not covered elsewhere in the TSRs that, if altered or modified, would have a significant effect on safety. Design Features are normally permanently built-in features that do not require or infrequently require maintenance or surveillance and are
normally not subject to change by operations personnel. The categories of Design Features to be addressed in accordance with DOE Order 5480.22 include the following:

- Vital passive components such as piping, vessels, supports, confinement structures, and containers
- Configuration and physical arrangement of the facility where safety is a concern including site characteristics such as the locations of public access roads, collocated facilities, facility area boundaries, site boundaries, and distances to the nearest residences
- Building materials if the safe operation of the facility depends on any component being constructed of a particular material

Changes to Design Features are considered significant modifications. The unreviewed safety question process ensures that changes to Design Features are appropriately analyzed and controlled so that they do not adversely affect safe operation of the facility. The configuration management system that controls changes to Design Features is discussed in Chapter 17.0, "Management, Organization and Institutional Safety Provisions."

Design Features for each of the facilities that, if altered or modified, would have a significant effect on safe operation are identified in the facility FSAR Annexes. Descriptions of these Design Features are provided in Chapter 2.0, "Facility Description." The safety functions they perform are provided in Chapter 4.0, "Safety Structures, Systems and Components."

5.7 INTERFACES WITH TECHNICAL SAFETY REQUIREMENTS FROM OTHER FACILITIES

The SNF Project facilities interface with each other and separate Hanford Site facilities both physically and administratively. These interfaces include utilities, fire protection, emergency preparedness, waste management, and SNF material transfers. SNF Project programs will be put in place to ensure that interactions with other facilities and their safety requirements do not affect the SNF Project facilities safety basis and do not exceed the safety analyses requirements.

The following discussion includes a description of the utilities, fire protection, emergency preparedness, and waste management interfaces and applicable safety programs. The SNF material transfers section includes a description of the transfer interfaces and TSRs required between SNF Project facilities the material is being transferred to or received from.
571 Utilities

The interfacing utilities include water and electrical power. The service water distribution system is the source of water for the Cold Vacuum Drying Facility (CVDF) and the Canister Storage Building (CSB) fire protection systems. Electrical power to the SNF Project facilities is supplied by the Hanford Site commercial power distribution system.

There are no utility interface requirements that affect the SNF Project facilities safety basis.

572 Fire Protection

The SNF Project fire protection program discussed in Chapter 11 interfaces with the Hanford Fire Department DOE-RL, and the Emergency Preparedness Program, discussed in Chapter 15. The fire protection program is established to meet the objectives of DOE Order 5480.7A, Fire Protection, to minimize the potential for a fire, ensure that fire does not cause an onsite or offsite release of radiological and other hazardous material that would threaten public health or the environment, provide an acceptable level of personnel life safety from fire, protect process control and safety systems from fire, ensure vital DOE programs do not experience unacceptable delays, and minimize property damage. The fire protection program is implemented through the Hanford Site policies and procedures on fire protection. An overview of the fire protection program is provided in HNF-PRO-340, Fire Protection Program Overview.

There are no fire protection interface requirements that affect the SNF Project safety basis.

573 Emergency Preparedness

The scope of the emergency preparedness program (see Chapter 15) ensures that appropriate actions are taken to protect the health and safety of the workers, the public, and the environment. Personnel at each facility are responsible for responding to all emergencies involving their facility and activities. The interfaces between the SNF Project facilities and other emergency organizations are described in DOE/RL-94-02, Hanford Emergency Response Plan, which applies to all Hanford Site employees and facilities. Under this plan, the contractor emergency response organization and the DOE-RL emergency response organization provide guidance to SNF Project facility personnel and oversee the response actions. DOE-RL interfaces with onsite emergency response organizations and offsite agencies including the DOE-Headquarters emergency management team.

There are no emergency preparedness interface requirements that affect the SNF Project facilities safety basis.
5.7.4 Facility Waste Management Program

The SNF Project waste management program (see Chapter 9.0) focuses on (1) protecting the workers, the public, and the environment, (2) ensuring proper management of waste from its point of generation to its final disposition, (3) ensuring compliance with applicable federal and state regulations, and (4) ensuring that radioactive and hazardous material exposure is as low as reasonably achievable. To accomplish these objectives, the waste management program contains the following basic elements:

- Plans and procedures governing waste designation, segregation, packaging, monitoring, treatment, storage, transportation, disposal, recordkeeping, and reporting
- Policies and programs for employee training, audits, and inspections
- Waste minimization goals and practices that reduce the volume of radioactive and hazardous waste and the spread of contamination

All waste transfers from the SNF Project facilities are controlled by procedures that meet the requirements specified in the appropriate DOE orders, Hanford Site requirements, federal and state regulations, and industry standards. WHC-EP-0063-04 Hanford Site Solid Waste Acceptance Criteria also identifies requirements for radioactive wastes and dangerous wastes as defined in WAC 173-303 "Dangerous Waste Regulations."

There are no waste management interface requirements that affect the SNF Project facilities safety basis.

5.7.5 Receipt and Transfer of Spent Nuclear Fuel

The SNF Project interfaces between the facilities through the receipt of SNF at CVDF from the K Basins and the transfer of the dried fuel from the CVDF to the CSB. This section describes the SNF material transfers between the K Basins, CVDF, and CSB, and applicable controls.

5.7.5.1 Spent Nuclear Fuel Transfers from K Basins to the Cold Vacuum Drying Facility
The K Basins send SNF to the CVDF that meets the criteria and assumptions identified in Chapter B3.0 of Annex B, the CVDF FSAR. The K Basins will send SNF to the CVDF in accordance with established controls that require a program to maintain consistency between the K Basins TSRs and the TSRs of interfacing facilities through the use of approved and controlled operating specifications and procedures. WHC-SD-WM-SAR-062 K Basins Safety Analysis Report does the following:

- Establishes maximum shipping temperatures to ensure the initial conditions and assumptions used in the CVDF safety analysis are valid.
• Establishes MCO basket loading controls and overall MCO loading controls to ensure the initial conditions and assumptions used in the CVDF and CSB safety analyses are valid.

• Establishes requirements for shipping preparations (such as cask-MCO void space purging requirements) to ensure the initial conditions and assumptions used in the CVDF safety analysis are valid.

• Defines the initiation of a shipping window to ensure that transfers from K Basins to the CVDF are adequately controlled such that the receipt pressure assumptions within the CVDF accident analyses are protected (Annex B, the CVDF FSAR, defines the completion point of the shipping window).

5.7.5.2 Spent Nuclear Fuel Transfers from the Cold Vacuum Drying Facility to the Canister Storage Building  The CVDF sends SNF to the CSB that meets the criteria and assumptions identified in Chapter A3.0 of Annex A the CSB FSAR. The only parameters of concern identified in Annex A are MCO seal leakage rates and maximum bulk water content. The CVDF will send SNF to the CSB in accordance with established controls that require a program to maintain consistency between the CVDF TSRs and the TSRs of interfacing facilities through the use of approved and controlled operating specifications and procedures. Annex B, the CVDF FSAR, does the following:

• Requires that proper testing be conducted to establish that the MCO water content is within acceptable levels to ensure that key parameters assumed in the CSB safety analysis are protected.

• Establishes a maximum MCO leak rate to protect leak rate assumptions in the CSB safety analysis.

5.8 REFERENCES


DOE Order 5480 22, Technical Safety Requirements  U.S. Department of Energy
Washington D.C.

DOE Order 5480 23  Nuclear Safety Analysis Reports, U S Department of Energy
Washington D.C.

DOE/RL-94-02, 1995 Hanford Emergency Response Plan  Rev 1, U S Department of Energy,
Richland Operations Office  Richland Washington

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CHAPTER 6.0

PREVENTION OF INADVERTENT CRITICALITY
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<th>Description</th>
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<tr>
<td>CCC</td>
<td>core component container</td>
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<tr>
<td>CPS</td>
<td>criticality prevention specification</td>
</tr>
<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
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<tr>
<td>CSER</td>
<td>criticality safety evaluation report</td>
</tr>
<tr>
<td>CSR</td>
<td>criticality safety representative</td>
</tr>
<tr>
<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
</tr>
<tr>
<td>FFTF</td>
<td>Fast Flux Test Facility</td>
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<tr>
<td>FRS</td>
<td>fuel retrieval system</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
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<tr>
<td>IAA</td>
<td>integrated audit and appraisal</td>
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<td>ID69</td>
<td>Ident-69</td>
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<tr>
<td>ISA</td>
<td>Interim Storage Area</td>
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<td>ISC</td>
<td>interim storage cask</td>
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<tr>
<td>IWTS</td>
<td>integrated water treatment system</td>
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<td>IXM</td>
<td>ion exchange module</td>
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<td>LWR</td>
<td>light water reactor</td>
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<td>MCO</td>
<td>multi-canister overpack</td>
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<tr>
<td>NAC</td>
<td>Nuclear Assurance Corporation</td>
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<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
</tr>
<tr>
<td>TRIGA</td>
<td>Test Reactor and Isotope Production General Atomics</td>
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6.0 PREVENTION OF INADVERTENT CRITICALITY

6.1 INTRODUCTION

This chapter provides the information necessary to document the Spent Nuclear Fuel (SNF) Project criticality prevention program. The material presented in this chapter is intended to describe the essential features of the inadvertent criticality prevention program as they relate to facility safety and to demonstrate its compliance with DOE Order 5480.24 Nuclear Criticality Safety. The SNF Project facilities (K Basins Cold Vacuum Drying Facility [CVDF], the Canister Storage Building [CSB], and the Interim Storage Area [ISA]) are classified as hazard category 2 facilities and have sufficient fissile materials to require a criticality prevention program.

In preparing the evaluations a graded approach was taken that factors in the low enrichment, storage configuration, and inventory limits of the stored N Reactor fuel. N Reactor fuel has a maximum initial enrichment of 1.25 wt% $^{235}$U, and other initial enrichments are 0.95 wt% $^{235}$U and 0.71 wt% $^{235}$U. The storage, handling, decapping, and repackaging of N Reactor fuel into shipping containers in the K Basins have been evaluated for potential nuclear criticality accidents and found to be safe and within the limits established for the fuel storage basins to meet nuclear criticality safety criteria (WHC-SD-WM-SAR-062). Fuel processing at the CVDF and long-term interim fuel storage at the CSB have also been determined to be safe and within the limits established to meet nuclear criticality safety criteria (see Annex B, the CVDF final safety analysis report [FSAR] and Annex A, the CSB FSAR). The buildup of fissile material in sludge, ion exchange modules, cartridge and sand filters, and K East Basin sandfilter backwash pits has been evaluated and the fissile material configurations were found to be critically favorable for all postulated event sequences and material arrangements (WHC-SD-WM-SAR-062).

Transportation of loaded multi-canister overpacks (MCOs) is discussed in HNF-SD-TP-SARP-017 Safety Analysis Report for Packaging Onsite Multi-Canister Overpack Cask. The single pass reactor fuel stored at the K Basins is currently not authorized to be processed at the CVDF or stored in the CSB.

The ISA is used to store sealed casks of SNF. Four types of SNF will be stored at the ISA.

- Fast Flux Test Facility (FFTF) fuel
- Test Reactor and Isotope Production General Atomics (TRIGA)$^1$ reactor fuel
- Pressurized water reactor fuel
- Boiling water reactor fuel

The storage and handling of the FFTF, TRIGA, and light water reactor (LWR) fuels at the ISA has been evaluated for potential nuclear criticality accidents and determined to be safe and within the limits established to meet nuclear criticality safety criteria (see HNF-3553 Annex D). Transportation of the storage casks to the ISA is discussed in WHC-SD-TP-SARP-008, Safety.

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$^1$TRIGA is a registered trademark of Gulf General Atomcs Company, Incorporated.
Analysis Report for Packaging (Onsite) NRF TRIGA Packaging, and WHC-SD-TP-SARP-010, Safety Analysis Report for Packaging (Onsite) Interim Storage Cask

See the facility FSAR Annexes for specific evaluations regarding the potential for criticality and the associated engineering and administrative controls

6.2 REQUIREMENTS

The requirements for SNF Project facility criticality prevention are listed in HNF-SD-SNF-RD-001, Spent Nuclear Fuel Project Standards/Requirements Identification Document. The specific requirements applicable to this chapter include:

- Title 10, Code of Federal Regulations, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR 72), specifically Section 72 124, "Criteria for Nuclear Criticality Safety"
- DOE Order 5480 24, Nuclear Criticality Safety
- DOE Order 5480 20A, Personnel Selection Qualification and Training Requirements for DOE Nuclear Facilities
- DOE Order 6430 1A, General Design Criteria
- ANSI/ANS-8 1-1983 Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors
- ANSI/ANS-8 3-1997 Criticality Accident Alarm System

6.3 CRITICALITY CONCERNS

6.3.1 Criticality Hazards

The criticality hazards discussed in this section involve the fissionable materials in the N Reactor fuel and fuel scrap that is stored in the Hanford Site K Basins. The SNF that is loaded in the MCOs may be in the form of intact fuel elements, partial fuel elements, or pieces of fuel scrap larger than 0.25 in. In addition, these fuel forms may contain fuel oxides and other corrosion products. A description of the N Reactor fuel elements is provided in Table 6-1. The types and quantities of N Reactor fuel analyzed for criticality concerns are described in...
Table 6-1  N Reactor Fuel Element Description

<table>
<thead>
<tr>
<th>Outer tube diameters</th>
<th>Mark IV fuel assembly (cm)</th>
<th>Mark IV fuel assembly (in)</th>
<th>Mark IA fuel assembly (cm)</th>
<th>Mark IA fuel assembly (in)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Zirconium cladding, outer diameter</td>
<td>6 160</td>
<td>2 425</td>
<td>6 106</td>
<td>2 404</td>
</tr>
<tr>
<td>Uranium, outer diameter</td>
<td>6 032</td>
<td>2 375</td>
<td>5 979</td>
<td>2 354</td>
</tr>
<tr>
<td>Uranium, inner diameter</td>
<td>4 422</td>
<td>1 741</td>
<td>4 592</td>
<td>1 808</td>
</tr>
<tr>
<td>Zirconium cladding, inner diameter</td>
<td>4 321</td>
<td>1 701</td>
<td>4 481</td>
<td>1 764</td>
</tr>
<tr>
<td>Outer tube enrichment</td>
<td>(wt%)</td>
<td>(wt%)</td>
<td>(wt%)</td>
<td>(wt%)</td>
</tr>
<tr>
<td>$^{235}\text{U}$</td>
<td>0.9470$^a$</td>
<td>1.250</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{236}\text{U}$</td>
<td>0.0392</td>
<td>0.0392</td>
<td></td>
<td></td>
</tr>
<tr>
<td>$^{238}\text{U}$</td>
<td>99.0138</td>
<td>98.7108</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Inner tube diameters</td>
<td>(cm)</td>
<td>(in)</td>
<td>(cm)</td>
<td>(in)</td>
</tr>
<tr>
<td>Zirconium cladding, outer diameter</td>
<td>3 249</td>
<td>1 279</td>
<td>3 165</td>
<td>1 246</td>
</tr>
<tr>
<td>Uranium, outer diameter</td>
<td>3 096</td>
<td>1 219</td>
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<tr>
<td>Uranium, inner diameter</td>
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<td>0 480</td>
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<td>0 440</td>
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<tr>
<td>Inner tube enrichment</td>
<td>(wt%)</td>
<td>(wt%)</td>
<td>(wt%)</td>
<td>(wt%)</td>
</tr>
<tr>
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<tr>
<td>$^{236}\text{U}$</td>
<td>0.0392</td>
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<td></td>
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</tr>
<tr>
<td>$^{238}\text{U}$</td>
<td>99.0138</td>
<td>99.0138</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel assembly dimensions</td>
<td>(cm)</td>
<td>(in)</td>
<td>(cm)</td>
<td>(in)</td>
</tr>
<tr>
<td>Maximum length</td>
<td>66 294</td>
<td>26 10</td>
<td>53 035$^b$</td>
<td>20 88$^b$</td>
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<td>End cap thickness</td>
<td>0 483</td>
<td>0 19</td>
<td>0 483</td>
<td>0 19</td>
</tr>
<tr>
<td>Fuel assembly weight (mass of uranium)</td>
<td>(kg)</td>
<td>(lb)</td>
<td>(kg)</td>
<td>(lb)</td>
</tr>
<tr>
<td>Maximum weight</td>
<td>23 4</td>
<td>51 6</td>
<td>16 7</td>
<td>36 7</td>
</tr>
</tbody>
</table>

Some Mark IV fuel (designated Mark IVB) contains natural uranium (0.71 wt % $^{235}\text{U}$) in both the outer and inner tubes.

$^a$ Twelve Mark IA fuel assemblies are 66 294 cm (26 1 in) long.
Most fissionable materials involved were initially at one of two enrichments before irradiation: 0.95 wt% or 1.25 wt% $^{235}\text{U}$. Operations procedures at the K Basins ensure that the fuel is segregated by enrichment and loaded in accordance with the limits described in WHC-SD-WM-SAR-062, K Basins Safety Analysis Report. Existing floor sludge and any material removed from the fuel during the cleaning process are discussed in WHC-SD-WM-SAR-062.

The fuel stored at the ISA is contained in sealed casks. Fuel from the FFTF has the highest initial enrichment, 29.28 wt% plutonium with a minimum of 11.63 wt% $^{240}\text{Pu}$. All fuel is in the form of intact assemblies, fuel pins, or fuel elements, and there are no pieces of fuel or fuel scrap as with the N Reactor fuel. Chapter 2.0 of HNF-3553, Spent Nuclear Fuel Project Final Safety Analysis Report, Annex D, “200 Area Intermediate Storage Area Final Safety Analysis Report,” contains a description of these casks and the different fuel design characteristics.

**6.3.1.1 N Reactor Fuel** The vast majority of the 2,100 metric tons of irradiated fuel stored in the K Basins is from the N Reactor. N Reactor fuel has a tube-within-tube design and was fabricated in two basic fuel assembly types, designated Mark IV and Mark IA. Mark IV fuel assemblies have a preirradiation enrichment of 0.95 wt% $^{235}\text{U}$ in both inner and outer elements and a maximum uranium mass of 23.4 kg. The assemblies have an outside diameter of 2.43 in and a length of 17.4 to 26.1 in. Mark IA fuel assemblies have a preirradiation enrichment of 1.25 wt% $^{235}\text{U}$ in the outer element, 0.95 wt% $^{235}\text{U}$ in the inner element, and a maximum uranium mass of 16.7 kg. They have an outside diameter of 2.40 in and a length of 14.9 to 26.1 in. There are 12 Mark IA fuel assemblies that are 26.1 in long, and they will be loaded into a Mark IV basket according to the requirements specified in WHC-SD-WM-SAR-062.

Table 6-1 provides a detailed listing of the N Reactor fuel dimensions and weights. Only the enrichment of unirradiated fuel, before it was loaded in the N Reactor, is given in the table. Analyses performed for N Reactor fuel cover both intact fuel assemblies, inner and outer fuel elements, and pieces of fuel assemblies. If possible, these pieces of fuel assemblies will be assembled into composite assemblies and loaded into fuel baskets. The criticality analysis performed for the contingency of dropping an MCO in which the fuel in all of the MCO baskets rubblizes into optimal scrap, bounds any loading of composite or partial assemblies. Pieces of fuel assemblies that cannot be assembled into composite assemblies were modeled as scrap (see Section 6.3.1.2).

Analyses on the effects of burnup and fission products decay showed that the unirradiated N Reactor fuel is neutronically more reactive (higher infinite criticality factor $k_\infty$) than irradiated fuel in spite of the presence of plutonium products in the SNF (HNF-SD-SNF-CSER-005). Reduced uranium enrichment and the presence of decayed or stable fission products more than compensate for any increase in $k_\infty$ attributable to the plutonium content in the SNF. Analysis of the effect on $k_\infty$ of fission product decay over a long period of time (e.g., 100 years or more) is described in HNF-SD-SNF-CSER-005, Criticality Safety Evaluation Report for the...
*Multi-Canister Overpack* and provides further justification for use of the unirradiated fuel characteristics in the analyses presented in this chapter. Because most of the fuel stored in the K Basins is irradiated, this is conservative for criticality analysis. Only 0.072% of the fuel is unirradiated.

Also stored in the K Basins are fuel types Mark IB, IC, IVB, and IVC. Mark IB and IVB have preirradiation enrichments of 0.71 wt% $^{235}$U and Mark IC has a preirradiation enrichment of 0.95 wt% $^{235}$U. There are no special handling requirements for these fuel types. The Mark IVC fuel has a preirradiation enrichment of 1.15 wt% $^{235}$U in the outer element and 0.95 wt% $^{235}$U in the inner element. No Mark IVC fuel is currently stored in the K Basins.

### 6.3.2 Reactor Fuel Scrap

The SNF that is to be shipped to the CSB is stored in the K East and K West Basins. A significant fraction of the fuel assemblies are assumed to be in a highly damaged or corroded condition, resulting in sludge and broken elements in the canisters. Canister sludge and fine particles of broken fuel are removed during the cleaning cycle of the fuel retrieval operation and are flushed to the integrated water treatment system. Any particulate material that is left after cleaning is classified as fuel scrap for the purpose of criticality analyses and will be loaded into the scrap baskets. It also is anticipated that removal from existing canisters, cleaning, and repacking of seemingly intact fuel assemblies in the K Basins could generate an additional amount of scrap fuel. Particulate that comes off of the fuel as it is dumped on the fuel retrieval system table will be swept into a scrap basket. The fuel will then be sorted. Broken pieces of the fuel assemblies that cannot be assembled into composite assemblies will be treated as scrap and packed separately in baskets specifically designed for loading scrap into the MCOs, as depicted in Figure 6-1. The Mark IA scrap basket has a center post similar to that of a Mark IA fuel basket, shown in Figure 6-2.

A region around the center of both scrap basket types is provided to collect fine pieces of scrap (this region is around the center post of the Mark IA scrap basket). Because the scrap was modeled as optimally sized spaced and moderated, the criticality analyses performed are bounding for all scrap loadings. Mark IV fuel assemblies and scrap are in the K West and K East Basins. A small amount of Mark IA fuel and scrap is stored in the K East Basin and the majority is stored in the K West Basin. Fine particulate material that falls through the perforated base plates of both the fuel and scrap baskets may build up on the floor of the basins. The scrap basket loading limits are based on analysis of spills of optimized scrap to the basin floor over a layer of this fine particulate. The fine particulate also was modeled as optimized scrap.

### 6.3.3 Suspended Uranium Dioxide

While in storage and after being loaded into the MCO, the SNF corrodes forming $\text{UO}_2$. Some of this $\text{UO}_2$ could be removed with the water as small suspended particles. This has been modeled as $\text{UO}_2$ enriched to 1.25 wt% $^{235}$U suspended in water, optimally moderated and reflected (HNF-SD-SNF-CSER-006). The results of the analysis are discussed in Chapter B6.0 of the CVDF FSAR (HNF-3553 Annex B).

### 6.3.4 Fast Flux Test Facility Fuel

Fuel consists of both standard driver fuel assemblies and test assemblies used to test both materials and fuel forms. All of the driver fuel assemblies and most of the test assemblies contain mixed oxide fuel containing $\text{UO}_2$ and $\text{PuO}_2$ in pellet form and...
clad with stainless steel or other experimental steel cladding. Slight variations in fuel enrichment exist but all are bounded by the maximum enrichment of 29.28 wt% plutonium with a minimum enrichment of 11.63 wt% $^{239}$Pu. The uranium in the mixed oxide fuel is either natural or depleted $^{235}$U. Many of the test assemblies were disassembled for post-irradiation testing. The fuel pins from these assemblies are contained in Ident-69 (ID69) pin containers. ID69 is the print designation for pin containers that fit in various locations in the interim examination and maintenance cell at the FFTF, there is more than one type of ID69 container. All SNF from the FFTF is contained in interim storage casks (ISCs) that will be shipped to the ISA. Included in the FFTF inventory are several test assemblies containing sodium-bonded uranium–zirconium metal fuel. This fuel is currently not authorized for storage at the ISA.

6.3.5 Test Reactor and Isotope Production General Atomics Fuel

TRIGA fuel is a ceramic zirconium hydride–uranium fuel with 8.5 wt% uranium content enriched to 20 wt% $^{235}$U. Elements are clad with either stainless steel or aluminum. Two fuel follower control rods also are to be stored at the ISA. The $^{235}$U content is listed in HNF-3553, Annex D, Table D2-4. The fuel is contained in Neutron Radiography Facility TRIGA casks and the fuel follower control rods are contained in U.S. Department of Transportation-6M casks. All these casks are contained in a Rad-Vault described in HNF-3553, Annex D, Section D2.3.

6.3.6 Commercial Light Water Reactor Fuel

The commercial LWR fuel consists of pressurized water reactor fuel assemblies and some boiling water reactor fuel pins which are zircaloy-clad $\text{UO}_2$ fuel enriched to a maximum of 3.6 wt% $^{235}$U. Fuel assemblies from the Point Beach and Calvert Cliffs facilities also were examined at Hanford. This fuel is stored in Nuclear Assurance Corporation (NAC)-1 casks.

6.3.2 Criticality Analysis

Table 6-2 shows the criticality issues for each facility. The specifics of the criticality analyses are discussed in the criticality safety evaluation reports written for SNF operations and are summarized in the facility FSAR Annexes.

An MCO is 166-in long with a 24-in outside diameter and a 0.5-in wall thickness. A total of five Mark IV fuel baskets or six Mark IA fuel baskets may be loaded into an MCO. The dimensions of the baskets are such that the total stack height of the baskets inside the MCO is about the same regardless of the type of basket. A shield plug assembly, consisting of a stainless steel plug, filters, and filter guard plate, is installed in the MCO after loading.

Rad-Vault is a trademark of Chem-Nuclear Systems, Incorporated.
<table>
<thead>
<tr>
<th>Facility</th>
<th>Operation</th>
<th>Criticality issue</th>
<th>FSAR Annex</th>
<th>Related CSER</th>
</tr>
</thead>
</table>
| CSB      | MCO storage | 1 MCO drop  
2 Receipt of flooded or misloaded MCOs  
3 Storage of flooded or misloaded MCOs  
4 Flooded storage tubes or sample station  
5 Flooded storage vault  
6 Misloaded storage tube (i.e. no intermediate impact absorber) | A | HNF SD SNF CSER 005 |
| CVDF     | MCO drying | Changes in moderation | B | HNF SD SNF CSER 005 |
|          | MCO water processing and storage | 1 Corrosion product buildup  
2 IXM/filter failures  
3 Spills  
4 Leaks  
5 Receiver tank overflows | B | HNF SD SNF CSER 006* |
| K Basins | Canister storage and movement | 1 Floor sludge  
2 Rack spacing  
3 Canister drops/spills  
4 Basin drain | K Basins FSAR | HNF SD SNF CSER 010* |
|          | Fuel retrieval and MCO basket loading | 1 Over batching in FRS equipment  
2 Sludge buildup in and around FRS equipment  
3 MCO basket spills  
4 FRS equipment failures  
5 Basin drain | K Basins FSAR | HNF SD SNF CSER 010* |
|          | MCO loading and movement | 1 MCO basket spills  
2 Misloaded MCO baskets  
3 Misloaded MCOs  
4 Dropped MCOs | K Basins FSAR | HNF SD SNF CSER 010*  
HNF SD SNF CSER 005* |
|          | Handling long length Mark IA assemblies in K West Basin  
Handling Mark IA material in K East Basin | Mark IA material in Mark IV MCO baskets | K Basins FSAR | HNF SD SNF CSER 010* |
| K Basins - IWTS | Fuel corrosion product or sludge handling | 1 Sludge accumulation in  
a Knockout pots  
b Settler tanks  
c Annular filters  
2 IWTS equipment failure  
3 Spills over the knockout pot storage array  
4 Sludge buildup around knockout pot array  
5 Basin drain | K Basins FSAR | HNF SD SNF CSER 011 |

*Related CSERs are marked with an asterisk (*).
Table 6-2  Criticality Issues of the Spent Nuclear Fuel Project Facilities (2 sheets)

<table>
<thead>
<tr>
<th>Facility</th>
<th>Operation</th>
<th>Criticality issue</th>
<th>FSAR Annex</th>
<th>Related CSER</th>
</tr>
</thead>
<tbody>
<tr>
<td>ISA</td>
<td>FFTF ISC storage</td>
<td>1 Flooding in ISC</td>
<td>D</td>
<td>WHC SD FF CSER 002^</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2 Flooding the ISA</td>
<td></td>
<td>WHC SD FF CSER 003^</td>
</tr>
<tr>
<td></td>
<td></td>
<td>3 ISC drop</td>
<td></td>
<td>WHC SD FF CSER 004^</td>
</tr>
<tr>
<td></td>
<td></td>
<td>4 Hot cell rot</td>
<td></td>
<td>WHC SD FF CSER 006^</td>
</tr>
<tr>
<td></td>
<td></td>
<td>5 ISC interaction</td>
<td></td>
<td>WHC SD SFV792 DA 004^</td>
</tr>
<tr>
<td></td>
<td></td>
<td>6 Seismic events</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>7 Misload ISC</td>
<td></td>
<td></td>
</tr>
<tr>
<td>TRIGA fuel storage</td>
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<td>1 Flooded cask</td>
<td>D</td>
<td>WHC SD FF CSER 006^</td>
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<tr>
<td></td>
<td></td>
<td>2 Tipped or dropped cask</td>
<td></td>
<td>WHC SD SQA CSA 30006^</td>
</tr>
<tr>
<td></td>
<td></td>
<td>3 Crushed cask</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Light water reactor fuel storage</td>
<td>Flooding</td>
<td>D</td>
<td>WHC SD FF CSER 007^</td>
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</tbody>
</table>

HNF SD SNF CSER 005 1999 *Criticality Safety Evaluation Report for the Multi Canister Overpack* Rev 4 Fluor Daniel Hanford Incorporated Richland Washington


4HNF SD SNF CSER 010 1998 *Criticality Safety Evaluation Report for the K Basin Fuel Retrieval Subproject* Rev 0 Fluor Daniel Hanford Incorporated Richland Washington


WHC SD FF CSER 003 1994 *Criticality Safety Evaluation for the Long Term Storage of FFTF Fuel in Interim Storage Containers* Rev 0 Westinghouse Hanford Company Richland Washington

1WHC SD FF CSER 004 1996 *Criticality Safety Evaluation for the Long Term Storage of FFTF Fuel in Interim Storage Casks* Rev 1 Westinghouse Hanford Company Richland Washington

WHC SD FF CSER 007 1998 *Criticality Safety Evaluation Report for TRIGA Fuel Storage at 400 Area Interim Storage Area* Rev 0 Westinghouse Hanford Company Richland Washington


WHC SD FF CSER 007 1996 *Criticality Evaluation for Long Term Storage of Light Water Reactor Fuel in NAC 1 Casks* Rev 0 Westinghouse Hanford Company Richland Washington


CSB = Canister Storage Building
CSER = criticality safety evaluation report
CVDF = Cold Vacuum Drying Facility
FFTF = Fast Flux Test Facility
FRS = fuel retrieval system
FSAR = final safety analysis report
ISA = Interim Storage Area
ISC = interim storage cask
IWTS = integrated water treatment system
IXM = ion exchange module
MCO = multi canister overpack
TRIGA = Test Reactor and Isotope Production General Atomics (fuel)
The normal condition is to have each type of SNF contained in baskets designated for that type of fuel (see Figures 6-1 and 6-2) and the baskets in turn stacked in an MCO (see Figure 6-3). The Mark IA fuel basket holds 48 Mark IA fuel assemblies and has a center post specifically designed for criticality control. This post is a hollow stainless steel bar with an outside diameter of 6.625 in and an inner diameter of 1.75 in to allow for the insertion of the long axial process tube for vacuum drying. The Mark IA scrap basket has a similar center post. The Mark IV fuel basket holds 54 fuel assemblies and has a central tube with a 2.64x1 outside diameter for installation of the long axial process tube to vacuum dry the fuel in the MCO at the CVDF. The designs of the MCO and fuel baskets are more fully summarized in HNF-SD-SNF-SARR-005, Multi-Canister Overpack Topical Report.

Each MCO may be loaded in one of three configurations:

- All fuel baskets
- One scrap basket with four or five fuel baskets
- Two scrap baskets and three or four fuel baskets

As discussed in WHC-SD-WM-SAR-062, the fuel basket loading is limited by the design of the fuel baskets. There is no minimum loading requirement for either Mark IA or Mark IV fuel assemblies. The Mark IA and Mark IV scrap baskets have administratively controlled maximum loading limits of 575 kg and 980 kg, respectively. These limits are discussed in WHC-SD-WM-SAR-062.

For criticality analyses of MCOs under normal conditions, each MCO was assumed to contain two scrap baskets, one at the top and one at the bottom, as shown in Figure 6-3, using the fuel basket loading with the highest $k_{eff}$. This loading was determined to contain partially loaded fuel baskets in the configurations shown in Figures 6-4 and 6-5. The Mark IA fuel basket contained 47 fuel assemblies with an empty location in the middle row, and the Mark IV basket contained 53 complete fuel assemblies with an inner element only in the outer row.

Each ISC contains a single core component container (CCC) that may hold a maximum of seven FFTF driver fuel assemblies or test assemblies. A CCC may only hold six ID69 pin containers because they are too long to fit in the central tube of the CCC. Several ISC loadings may be considered to be normal and these are discussed in HNF-3553 Annex D Section D6 3 3 1 1.

The normal loading for the TRIGA fuel cask contains 18 fuel elements, and the Rad-Vault contains six casks. Two U.S. Department of Transportation-6M casks, each containing a single fuel follower control rod, also may be loaded in the Rad-Vault. The center location of the Rad-Vault contains an empty 55-gal drum and the casks are loaded around the outer wall of the vault. Details of the models are discussed in HNF-3553 Annex D Section D6 3 3 2 1.

Similar to the ISCs, the NAC-1 casks have different loadings that may be considered to be normal. These are also discussed in HNF-3553 Annex D Section D6 3 3 3 1.

November 1999
6.3.3 Criticality Criteria and Assumptions

6.3.3.1 Criteria for Acceptable Criticality Determinations  The criticality prevention criterion is that the effective neutron multiplication factor ($k_{\text{eff}}$) shall not exceed $k_{\text{limit}}$ ($k_{\text{eff}} \leq 0.95$ for a loaded MCO) including allowances for all uncertainties and meeting all single contingencies. For criticality safety, the double contingency principle requires that process design incorporate sufficient factors of safety such that at least two unlikely independent, or concurrent changes occur before a criticality accident is possible. The analyses presented in the annexes to this FSAR demonstrate that the double contingency principle is satisfied by showing that the allowed configurations of fissile material in all components and systems will not exceed the $k_{\text{limit}}$ for the violation of any single contingency. The criterion that $k_{\text{limit}}$ not exceed 0.95 for all types and locations of storage casks is based on implementing the US Nuclear Regulatory Commission equivalency requirements described in HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward Additional NRC Requirements, and on the US Department of Energy’s policy that new SNF Project facilities achieve a level of nuclear safety equivalent to that of comparable facilities licensed by the US Nuclear Regulatory Commission (Sellers 1995). The criticality prevention criterion may be stated as follows:

\[ k_{\text{eff}} = k_{\text{calc}} - \Delta k_{\text{bias}} + (\sigma_b^2 + \sigma_c^2 + \sigma_d^2 + \sigma_e^2 + \sigma_T^2)^{1/2} \leq k_{\text{limit}} \]

where:

\[ k_{\text{eff}} = \text{calculated effective neutron multiplication constant corrected for bias and uncertainties} \]
\[ k_{\text{calc}} = \text{calculated } k_{\text{eff}} \]
\[ \Delta k_{\text{bias}} = \text{methods bias} \]
\[ \sigma_b = \text{bias uncertainty} \]
\[ \sigma_c = \text{calculational uncertainty} \]
\[ \sigma_d = \text{uncertainty in } k_{\text{eff}} \text{ due to dimensional tolerances} \]
\[ \sigma_e = \text{uncertainty in } k_{\text{eff}} \text{ due to enrichment tolerances} \]
\[ \sigma_T = \text{uncertainty in } k_{\text{eff}} \text{ due to temperature variations} \]

No uncertainty is calculated for variations in the position of the scrap basket in the MCO. This is because the scrap basket is modeled such that the scrap is as close as possible to the bottom of the fuel basket and the scrap basket sides are expanded as widely as possible to conserve mass. The dimensions used exceed the manufacturing tolerances for the basket.

The calculated $k_{\text{eff}}$ must be at the 95% confidence level. For a one-sided distribution, this level is obtained by multiplying the total statistical uncertainty by 1.645 except for $\sigma_b$, which is multiplied by a factor of 2. Applying the factors to get the upper bound of $k_{\text{eff}}$ at the 95% confidence level results in the following equation:

\[ k_{\text{eff}} = k_{\text{calc}} - \Delta k_{\text{bias}} + [(2\sigma_b)^2 + (1.645^2[\sigma_c^2 + \sigma_d^2 + \sigma_e^2 + \sigma_T^2])]^{1/2} \leq k_{\text{limit}} \]
The above formula is generic for all those represented in the facility FSAR Annexes. For all calculations involving SNF in the MCO, the values of the $k_{\text{eff}}$ equation parameters are as follows:

- $\Delta k_{\text{bus}} = -0.0004$ (HNF-SD-SNF-ANAL-013)
- $\sigma_b = 0.005$ (HNF-SD-SNF-ANAL-013)
- $\sigma_d = 0.000546$ (HNF-SD-SNF-CSER-005)
- $\sigma_c = 0.00154$ (HNF-SD-SNF-CSER-005)
- $\sigma_T = 0.00081$ (HNF-SD-SNF-CSER-005)

For a loaded MCO, the equation simplifies to:

$$k_{\text{eff}} = k_{\text{calc}} + 0.0004 + \left[ (0.01)^2 + (1.645)^2 (0.002083^2) \right]^{1/4} \leq 0.95$$

Calculations for the ISCs used the MONK6A and MONK6B codes. No uncertainties were calculated for parameter variations. Instead, fuel enrichments and component dimensions were used that give the largest $k_{\text{eff}}$ values. The equation in this case is as follows:

$$k_{\text{eff}} = k_{\text{calc}} - \Delta k_{\text{bus}} + (2\sigma_b)^2$$

This reduces to the following equation:

$$k_{\text{eff}} = k_{\text{calc}} + 0.015$$

where $(2\sigma_b)^2 - \Delta k_{\text{bus}} = 0.015$ (WHC-SD-SQA-CSWD-20015).

TRIGA fuel was analyzed using MCNP, MONK6B, and WIMS-E/GOLF codes. The application of the uncertainties and biases for each code is discussed in HNF-3553, Annex D, Section D6 3.3.2.1.

The calculations for the LWR fuel in the NAC-1 casks were performed with MCNP and KENO-Va/Seale 4.2. Section D6 3.3.1 of HNF-3553, Annex D, discusses the application of the uncertainties and biases for these analyses.

### 6.3.3.2 Analysis Assumptions for the N Reactor Fuel

All criticality analyses for the CSB CVDF and K Basins are based on conservative assumptions for determining the worst-case normal and credible accident conditions. These assumptions are listed below and are common to all the analyses:

- Optimal moderation for the scrap baskets
- Optimal scrap size (unless controlled)
- Optimal spacing
- Maximum enrichment
- Unirradiated fuel (no burnup)
The assumption of optimal moderation is applied to fuel whenever the fuel geometry is not constrained by the storage canisters MCO fuel baskets, or MCO structure. During abnormal events the fuel was also modeled as optimally moderated and spaced unless constrained by safety class components. If the fuel was assumed to rubblize, it was considered to be scrap. Fuel scrap and floor sludge were always assumed to be optimally configured. The sludge in the integrated water treatment system was modeled as being optimally moderated and spaced based on the maximum particle size of 550 μm. This size is controlled by the safety-class screens on the knockout pots.

6.3.3 Analysis Assumptions for the Fast Flux Test Facility Fuel. The criticality analysis for the FFTF fuel in the ISCs is based on the following set of assumptions:

- The fuel modeled is outer driver fuel, Type 41, with 29.28 wt% plutonium and 11.63 wt% 240Pu, which is the highest total plutonium enrichment for any of the FFTF fuel pins.
- Unirradiated fuel was modeled, no credit was taken for burnup.
- The radial dividers in the ID69 pin containers were neglected (resulting in a positive bias in $k_{eff}$).

A corrosion phenomenon that has been observed for nuclear fuel irradiated in a sodium-cooled reactor and exposed to the corrosive environment of cleaning the residual sodium from the assemblies is called hot-cell rot. This has been observed to occur at other facilities when cleaned assemblies have been stored for long periods of time. Hot-cell rot has not been observed with assemblies from FFTF. For the purpose of this analysis, hot-cell rot was assumed to occur and fuel pins and/or assembly ducts suffer a complete structural failure. Hot-cell rot was not considered credible for ID69 pin containers or CCCs.

6.3.4 Analysis Assumptions for the Test Reactor and Isotope Production General Atomics Fuel. Because different codes were used for the analysis of the TRIGA fuel, the assumptions are not presented in this section. HNF-3553 Annex D Section D6 3 3 2 1 discusses all the models used in the analysis.

6.3.5 Analysis Assumptions for the Light Water Reactor Fuel. The criticality analysis for the LWR fuel in the NAC-1 casks is based on the following set of assumptions:

- Unirradiated fuel was modeled; no credit was taken for burnup.
- The zircaloy cladding was neglected.
- Enrichment for all pins in the model was taken as the highest actual value for any single pin in the assembly.
The casks containing the Point Beach assemblies have a maximum of 179 pins with 3.2 wt% enrichment.

The cask containing the Calvert Cliffs full assembly has a maximum enrichment of 2.72 wt%.

The cask containing the Calvert Cliffs partial assembly has a maximum enrichment of 2.73 wt%.

6.4 CRITICALITY CONTROLS

The requirements for determining engineered and administrative criticality controls are provided by DOE Order 5480.24 and are implemented using the guidance of HNF-PRO-537, Criticality Safety Control of Fissionable Material.

6.4.1 Engineering Controls

The preferred method of ensuring criticality safety is to use passive engineered controls although active engineered controls are also acceptable.

- Engineered controls for the MCOs include features such as the 6 625-in. center post for the Mark IA fuel and scrap baskets and the internal diameter for the MCO.
- Engineered controls in the K Basins include the canisters and components in the fuel retrieval system.
- Engineered controls for the ISCs include both the ISC and the CCC, which are designed to maintain structural integrity for all design basis accidents.
- For the TRIGA fuel, the engineered features include the Neutron Radiography Facility cask, which may hold a maximum of eighteen fuel elements.
- The engineered control for the LWR fuel is the inner pipe containing the loose fuel pins in the NAC-1 cask.

See the facility FSAR Annexes for specific engineering controls.

Configuration control of equipment used to store, handle, transport, and process SNF is maintained in accordance with the guidance of HNF-PRO-544, Criticality Plant Configuration Control. A facility safety equipment list designates all equipment that may impact criticality safety as safety-class equipment. Before work on the equipment, a work authorization is processed, including review by the facility criticality safety representative (CSR) and approval by the facility.
safety organization Chapter 17 of the SNF Project FSAR provides a general description of the SNF Project configuration control program

6.4.2 Administrative Controls

Administrative controls in addition to engineering controls are implemented at each facility to ensure criticality safety. Types of controls are basket loading limits and MCO loading limits. The ISC has administrative controls on loading, and the administrative control for the LWR fuel requires loose fuel pins be loaded into the inner pipe. Administrative controls for the TRIGA fuel include limits on loading the Rad-Vault and spacing between the Rad-Vault and other fuel. See the facility FSAR Annexes for specific administrative controls.

Administrative control of processes and system configurations important to criticality safety is maintained in accordance with the guidance of HNF-PRO-544. Proposed changes to design or process configurations are reviewed by the CSR to ensure that such a change can be performed under an approved evaluation and criticality prevention specification. All changes that impact criticality safety are documented and include proper approval designation.

6.4.3 Application of Double Contingency Principle

The double contingency principle, as defined in DOE Order 5480.24 states that designs shall incorporate sufficient factors of safety to require at least two unlikely independent and concurrent changes in process conditions before a criticality accident is possible. DOE Order 6430.1A Chapter 13 Section 1320, delineates criteria for irradiated fissile material storage facilities. It provides no specific guidance on double contingency but references 10 CFR 72 for further guidance. The double contingency criterion delineated in paragraph 10 CFR 72.124 (a) is similar to the one quoted in DOE-STD-3009-94 Preparation Guide for US Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports.

The design and operation of each facility has been analyzed and engineered, and administrative controls implemented, to ensure the double contingency principle is satisfied. These controls are reported in the facility FSAR annexes. All facility FSAR Annexes purport to satisfy the double contingency requirement by ensuring that any single failure shall not result in criticality. For conservatism, the safety analyses were based on unirradiated fuel (initial enrichment) critical mass parameters. Normal operating conditions for each facility were identified. Potential error or failure situations were developed using the guidelines of Appendix A of HNF-PRO-539 Criticality Safety Evaluations and by evaluating the operations at each facility.
6.5 CRITICALITY PROTECTION PROGRAM

This section presents an overview of the organizational structure and interfaces and the technical and administrative practices of the criticality protection policy and programs for the SNF Project operations. An overview of the criticality safety management program for the SNF Project is shown in Figure 6-6. Each facility is responsible for developing its own criticality safety program in accordance with the requirements of DOE Order 5480.24

6.5.1 Criticality Safety Organization

The organizational responsibilities for criticality safety are described in HNF-PRO-334, Criticality Safety General Requirements. The procedure identifies the organizational elements responsible for individual aspects of the criticality safety and defines the responsibilities of the overall facility management and the operators who handle the fissionable material (i.e., the SNF). The procedure also defines the responsibilities of the engineering organization that performs criticality analyses to establish criticality safety parameters for each individual facility. The SNF Project Safety Review Board provides oversight of all criticality matters. The organization and its responsibilities are described in Chapter 11.0 of the SNF Project FSAR.

The CSR conducts periodic facility inspections and reviews modifications to operating procedures and facility equipment for impact on criticality safety. The CSR also provides information for or conducts the required criticality safety meetings for the operators, prepares criticality prevention specifications and postings, and serves as a point of contact for all criticality safety-related issues.

6.5.2 Criticality Safety Plans and Procedures

The criticality safety plans and procedures for SNF Project operations were developed as required by DOE Order 5480.24. Each facility develops its procedures to comply with the above plans. The general process for preparation, review, approval, distribution, and use of SNF Project procedures is described in Chapter 12.0 of the SNF Project FSAR.

Included in the procedures are provisions for posting criticality safety limits, material and operational controls, review of operations, emergency evacuation, and guidelines for permitting fire fighting water or other moderating materials used to suppress fires within or adjacent to moderation control areas. The general fire fighting provisions are described in Chapter 11.0 of the SNF Project FSAR.

6.5.3 Criticality Safety Training

Criticality safety training requirements are provided in DOE Order 5480.24. The guidance of HNF-PRO-538, Criticality Safety Training, has been used to develop a training program for
the SNF Project operations organization. This program ensures that operators, supervisors, and managers involved in the control, handling, storage, or transfer of the SNF in the MCOs understand their responsibilities for preventing criticalities. The training program ensures that the knowledge and experience profile of all management personnel is commensurate with their assigned responsibilities and that they are aware of the criticality risks involved and the preventive and corrective actions required in performing their responsibilities. The general process for training SNF Project workers is described in Chapter 120 of the SNF Project FSAR.

The Hanford Site training organization provides formal criticality safety training classes for fissile material handlers and managers and engineers. This formal classroom training, which includes written examinations with a minimum passing grade, is required every two years. General topics covered in the formal classroom training include the following:

- Basic criticality principles (consequences of a criticality accident, safety factors for \( k_{\text{eff}} \), and responsibilities for contingency prevention)
- Labeling and posting
- Emergency procedures and required responses

On-the-job facility-specific continuing training is provided through periodic safety meetings as considered necessary by the supervisors and managers with the guidance of HNF-PRO-538. As a part of this training, the facility CSR provides quarterly criticality safety training information to the managers and operators for their review. Annually, the CSR conducts a safety meeting for managers and operators to review this criticality safety training information and present new material as appropriate. Time is allowed during the facility-specific training for the discussion of unusual occurrences and work-oriented problems. Changes to procedures and process standards (which contain the criticality prevention specifications), technical safety requirements, management requirements, and procedures, and emergency procedures, also are addressed.

In addition to the periodic training sessions, each manager of personnel who handle SNF packages is responsible for on-the-job training of those personnel. The workers are not only trained in the use of facility-related procedures and process standards but also in the actual use of equipment to perform the various jobs.

Formal training records that verify satisfactory completion of formal training are maintained by the training organization. The records for job-specific orientation and periodic training are established and maintained by facility management.

### 6.5.4 Determination of Operational Nuclear Criticality Limits

For SNF Project facilities, HNF-SD-SNF-DB-003, in accordance with U.S. Department of Energy policy for U.S. Nuclear Regulatory Commission equivalency (Sellers 1995) requires that \( k_{\text{eff}} \) shall not exceed 0.95 for the CSB, CVDF, and MCO loading (0.98 for SNF in the K Basins).
until loaded into an MCO) The $k_{\text{eff}}$ shall not exceed 0.95 at the ISA. These calculational $k_{\text{eff}}$ limits provide an adequate margin of safety with respect to a criticality $k_{\text{eff}}$ of 1.0.

Criticality safety evaluations are performed in accordance with the guidance of HNF-PRO-537. Unless otherwise stated, all calculations used to verify compliance with the limits were performed using the Monte Carlo computer code MCNP (LA-12625). MCNP is used internationally and has been extensively tested with the ENDF/B-V-based cross sections. Use of other verified and validated codes to verify compliance will be discussed in the affected criticality safety evaluation reports.

The bias and uncertainty for MCNP calculations of low-enriched uranium systems were derived in HNF-SD-SNF-ANAL-013 *MCNP4B Criticality Validation and Bias for Low-Enriched Uranium Systems* and the uncertainties due to dimensional and enrichment tolerances were derived in HNF-SD-SNF-CSER-005. As described in Chapters 6.0 of the facility FSAR Annexes, in HNF-SD-SNF-CSER-005, the calculated values of $k_{\text{eff}}$ are corrected for biases in MCNP and for the dimensional and material uncertainties. WIMS-E validation and MONK6A and 6B validations are described in WHC-IP-0840-FMEF *Validation of WIMS-E for Prediction of Uranium Plutonium Nitrate Solution Critical Masses*, WHC-SD-SQA-CSWD-20015, *MONK6A Pu Validation* and WHC-SD-SQA-CWSD-20019 *MONK6B Pu Validation*. The application of biases and uncertainties is discussed in HNF-3553, Annex D, Section D6.0.

Analyses also are performed to account for operator error contingencies, such as loading Mark 1A material into a Mark IV basket or misloading ISCs, to ensure that the criticality criteria are not violated by those contingencies thus meeting the requirements of DOE Order 5480 23, *Nuclear Safety Analysis Reports* paragraph 4 f(3)(d)& item a, d and e. The limits are established by the criticality safety evaluation reports which are used to develop the criticality specifications that are identified in the technical safety requirements document for each of the SNF Project facilities.

### 6.5.5 Criticality Safety Inspections and Audits

Criticality safety inspections, audits, and assessments are performed at the SNF Project facilities in accordance with the guidance of HNF-PRO-548 *Criticality Safety Inspections and Assessments*. The SNF Project nuclear safety organization periodically verifies and validates that the facility is implementing the criticality safety program adequately. The current DOE Hanford Site contractor Nuclear Safety organization and the Criticality and Shielding organization conduct independent inspections, and make recommendations for program improvements for all Hanford Site nuclear facilities as shown in Figure 6-6 for the SNF Project.

Criticality safety inspections are performed routinely at the facility. The CSR performs monthly or more frequent inspections to verify correct fuel storage thus meeting the requirement of DOE Order 5480 23 paragraph 4 f(3)(d)&, item f and the guidance of HNF-PRO-548.
The criticality safety inspection and assessment reports are maintained in accordance with the SNF Project records management program described in Chapter 17.0 of the SNF Project FSAR. These records are available for review during assessments and also for review of trending for nonconformances.

6.5.6 Criticality Infraction Reporting and Follow-Up

The criticality infraction reporting and follow-up program provides for safe recovery from criticality safety program nonconformances. The underlying principle of the program is to not make a nonconforming situation worse and to prevent recurrence. The nonconforming situation is stabilized as necessary notifications are made, and following evaluation a recovery plan may be developed, approved, and implemented. A long-term corrective action plan may be developed and lessons learned may be incorporated into the facility safety analysis. The nonconformance actions are documented and the records retained in accordance with the SNF project records management program.

The manager of SNF Project Operations is responsible for developing procedures on reporting and follow-up of criticality infractions at SNF Project facilities. The procedures are developed in accordance with the guidelines provided in HNF-PRO-549 “Criticality Safety Nonconformance Response” and HNF-PRO-060 “Reporting Occurrences and Processing Operations Information.”

6.6 CRITICALITY INSTRUMENTATION

Criticality alarm and detection systems are not required for any of the SNF Project facilities. Each facility FSAR Annex contains the specific discussions relating to criticality instrumentation requirements and justification for not requiring criticality alarm and detection systems.

6.7 REFERENCES


DOE Order 5480 20A, Personnel Selection Qualification and Training Requirements for DOE Nuclear Facilities, U S Department of Energy, Washington, D C

DOE Order 5480 23, Nuclear Safety Analysis Reports U S Department of Energy, Washington, D C

DOE Order 5480 24, Nuclear Criticality Safety, U S Department of Energy, Washington, D C

DOE Order 6430 1A, General Design Criteria U S Department of Energy, Washington, D C


HNF-PRO-060, Reporting Occurrences and Processing Operations Information Project Hanford Policy and Procedure Fluor Daniel Hanford Inc Richland Washington


HNF-PRO-537, Criticality Safety Control of Fissileable Material, Project Hanford Policy and Procedure (PHPP) System, Fluor Daniel Hanford, Inc Richland, Washington


HNF-PRO-539 Criticality Safety Evaluations, Fluor Daniel Hanford, Incorporated, Richland, Washington

HNF-PRO-544 Criticality Plant Configuration Control Fluor Daniel Hanford, Incorporated Richland Washington

HNF-PRO-548 Criticality Inspections and Assessments Project Hanford Policy and Procedure (PHPP) System, Fluor Daniel Hanford, Inc, Richland, Washington

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Figure 6-1  Fuel Scrap Storage Basket
Figure 6-2 Fuel Assemblies Storage Basket

Mark IA

Mark IV

Criticality control insert
Top of Fuel Rod
Steel Baseplate (typical)
Crane Grapple Adapter (Inside)
Solid Jacket (typical)

-1.5 m
21.9 m

22.625 in (maximum)

-11 in
27.8 in

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Figure 6-3  Normal Basket Loadings in Multi-Canister Overpacks Used in the Models
Figure 6-4 Maximum Reactivity Loading Arrangement for Mark IA Fuel in Multi-Canister Overpack
Figure 6-5  Maximum Reactivity Loading Arrangement for Mark IV Fuel in Multi-Canister Overpack
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RADIATION PROTECTION
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LIST OF TERMS

ALARA as low as reasonably achievable
CSB Canister Storage Building
CVDF Cold Vacuum Drying Facility
DAC derived air concentration
DOE U S Department of Energy
dpm disintegrations per minute
FSAR final safety analysis report
MCO multi-canister overpack
NRC U S Nuclear Regulatory Commission
PHMS Project Hanford Management System
RWP radiological work permit
SNF spent nuclear fuel
S/RID standards/requirements identification document
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7 0 RADIATION PROTECTION

7 1 INTRODUCTION

This chapter describes the safety policies, procedures, designs, and other considerations relative to radiation protection that maintain radiation exposures as low as reasonably achievable (ALARA) at the Spent Nuclear Fuel (SNF) Project facilities. Credible radiological hazards are identified in Chapters 30 of the facility annexes to the SNF Project Final Safety Analysis Report (FSAR) and are not detailed in this chapter. The radiation protection program elements described in this chapter are designed to minimize potential exposure from radiological accidents and to minimize occupational exposures resulting from normal and off-normal operations.

In those cases where policies, programs, and practices important to safe operation are described in detail in other documents, the information is summarized in this chapter and the documents are referenced. The detailed programs and procedures described in referenced documents may be changed without further U.S. Department of Energy (DOE) approval to the extent that the changes do not constitute an unreviewed safety question as defined by DOE Order 5480.21, Unreviewed Safety Questions.

7 2 REQUIREMENTS

The requirements that form the basis for the radiation protection program are found in HNF-SD-SNF-RD-001 Spent Nuclear Fuel Project Standards/Requirements Identification Document. The specific requirements applicable to this chapter include:

- Title 10 Code of Federal Regulations Part 835, "Occupational Radiation Protection" (10 CFR 835)

7 3 RADIATION PROTECTION PROGRAM AND ORGANIZATION

7 3 1 Radiation Protection Program Elements

The radiation protection program for the SNF Project protects against the basic types of radiological hazards including the following that could affect occupational exposure:

- Contamination with radioactive material
- Ionizing radiation emitted from the multi-canister overpack (MCO) pipes, equipment, or other radiation sources

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Potential releases of radioactive aerosols, gases, or particles that could become airborne and affect occupational exposure, either by inhalation or external exposure.

Chapters 30 of the facility FSAR Annexes contain a summary of the hazards associated with the SNF Project facilities. The SNF Project radiation protection program is implemented in accordance with standards/requirements identification document (S/RID) Section 11 (HNF-SD-SNF-RD-001), 10 CFR 835, and HSRCM-1, *Hanford Site Radiological Control Manual* to protect against those hazards and against unnecessary radiation exposures from any radiation source in the SNF Project facilities. Table 7-1 identifies the primary elements of the radiation protection program.

### Table 7-1 Primary Program Elements for Radiation Protection

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*10 CFR 835 Occupational Radiation Protection Code of Federal Regulations as amended

ALARA = As low as reasonably achievable

### 7.3.2 Radiological Control Organization

Radiation protection organizational requirements are mandated by HSRCM-1. The requirements mandate that a radiological control manager be appointed who is responsible for overseeing a radiation protection program. Field work is performed by radiological control.
technicians supported by professional staff. Qualifications and staffing are consistent with the requirements contained in 10 CFR 835. Specific staffing assignments within the radiological control organization are subject to the discretion of the radiological control manager based on activities to support safe operation of SNF Project facilities.

7.4 ALARA POLICY AND PROGRAM

Minimizing radiation exposure to levels that are ALARA is a fundamental principle of radiation protection. 10 CFR 835.101, "Radiation Protection Programs," mandates formal plans and measures for applying the ALARA process in controlling occupational radiation exposure. To support a comprehensive ALARA program, a review of applicable U.S. Nuclear Regulatory Commission (NRC) documents was made by the SNF Project organization. As a result of this review, sections of NRC Regulatory Guide 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable, are to be applied in the Cold Vacuum Drying Facility (CVDF) and Canister Storage Building (CSB) designs as applicable and address piping systems that carry radioactive materials. Regulatory Guide 8.8 is listed as an additional requirement in HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements. Each facility FSAR Annex addresses this design requirement.

ALARA policy during facility operations and/or design is described in the following:

- HNF-PRO-1618, ALARA Policy and Management Commitment
- HNF-PRO-1619, ALARA Organization and Responsibilities
- HNF-PRO-1620, ALARA Program Scope
- HNF-PRO-1621, ALARA Decision-Making Methods
- HNF-PRO-1629, Administrative Control Levels
- HNF-PRO-1631, ALARA Training
- HNF-PRO-1633, ALARA Program Records
- HSRCM-1, Hanford Site Radiological Control Manual
- SNF Project administrative procedures

The radiation protection program governing facility activities follows these policies by implementing ALARA into all facets of radiological work. These policies specifically commit SNF Project organizations to limiting radiation exposures to ALARA levels and acknowledge that the primary method used to achieve ALARA objectives is physical design features. The SNF Project incorporated ALARA principles early on in the design of the CVDF and CSB and in the K Basins modifications.
741 Program Administration

Essential elements of an acceptable occupational ALARA program are defined in DOE G4441-1-2 *Occupational ALARA Program Guide for Use with Title 10 Code of Federal Regulations Part 835 Occupational Radiation Protection* The essential elements as implemented in the SNF Project ALARA program are defined as follows:

- Management commitment
- Assigning specific responsibilities
- Establishing challenging administrative control levels
- Establishing radiological performance goals
- Conducting ALARA training
- Establishing plans and procedures
- Performing internal reviews and audits
- Using optimization methodology to reduce exposure
- Performing radiological design reviews
- Performing radiological work planning
- Maintaining ALARA records

General criteria for an ALARA committee are detailed in HSRCM-1, Article 138 The ALARA program implemented at the SNF Project facilities is documented in SNF Project administrative procedures, which establish lines of responsibility for ensuring that ALARA practices are administered properly.

742 Radiological Design

7421 Requirements The basic elements of radiological (ALARA) design as required by 10 CFR 835 HSRCM-1, and HNF-SP-1145, *Fluor Daniel Hanford Radiation Protection Program, Implementation of Title 10 Code of Federal Regulations Part 835* are summarized below:

- Measures must be taken to keep radiation exposures ALARA by the combination of facility and equipment design and administrative controls The primary methods shall be physical design features, administrative controls shall be used only as supplemental methods to keep exposures ALARA (10 CFR 835 1001(a))

- Physical design features must be demonstrated to be impractical before administrative controls can be used to maintain radiation exposures ALARA (10 CFR 835 1001(b))

- Dose reduction, contamination reduction, and waste minimization features are to be incorporated into the design at the earliest planning stages (HSRCM-1)
Radiological design reviews must ensure the integration of appropriate methods and considerations during the design phase to maintain exposures ALARA during the construction, modification, and operation of the equipment and facility.

During the design phase, optimization methods must be used in developing and justifying facility design and physical controls to ensure that exposure is maintained ALARA during subsequent maintenance and operations (10 CFR 835 1002(a)).

"The design objective for controlling personnel exposure from external sources of radiation in areas of continuous occupational occupancy (2,000 hours per year) shall be to maintain exposure levels below an average of 0.5 mrem (5 microsieverts) per hour and as far below this average as is reasonably achievable" (10 CFR 835 1002(b)).

Actions must be taken to meet the design objective for exposure rates in areas not continuously occupied; this design objective is for exposure rates to be ALARA, and less than exposure rates that would result in an occupational worker receiving more than 20% of the applicable standard in 10 CFR 835 202, 'Occupational Exposure Limits for General Employees' (10 CFR 835 1002(b)).

Actions must be taken to meet the design objective for the control of airborne radioactive material; this design objective is to normally avoid the release of radioactive material to the workplace atmosphere, and in any case to control the inhalation of radioactive material to levels that are ALARA (10 CFR 835 1002(c)).

The design of the facility and the selection of materials shall include features that facilitate operations, maintenance, decontamination, and decommissioning (10 CFR 835 1002(d)).

7 4 2 2 Design  For facility-specific radiological design considerations see the facility FSAR Annexes. Radiological design considerations that affect exposures at all SNF Project facilities are described in Section 10 of HNF-SD-SNF-SARR-005 Multi-Canister Overpack Topical Report.

7 4 3 Performance Goals and Indicators

HSRCM-1 establishes methods for assessing the effectiveness of the radiological protection program including ALARA measures. Under this program, the SNF Project organization has established a system for defining and tracking specific radiological performance goals and performance indicators.
7.4.4 Radiological Work Planning

The planning of radiological work is a key element in the ALARA program. Radiological work at the SNF Project facilities will be planned in compliance with 10 CFR 835 requirements in accordance with HSRCM-1. Radiological activities are subject to planning steps (described in HSRCM-1, Articles 312 and 313) that include formal radiological review, senior management review, ALARA committee review, prejob briefing, enhanced management oversight, and post-job review.

7.4.5 Radiological Work Documentation and Reviews

SNF Project facilities integrate ALARA into the planning, conduct, and post-job reviews of all radiological work in compliance with 10 CFR 835 and HSRCM-1. Key features of this system are discussed in the following subsections.

7.4.5.1 Prejob ALARA Review
A prejob ALARA review ensures that necessary ALARA considerations have been included in the prejob planning to ensure that exposures are minimized. The ALARA Management Worksheet, or its equivalent, documents prejob ALARA considerations, ALARA practices, and the estimated collective dose. The trigger levels (HSRCM-1, Article 312) for deciding on the use of an ALARA Management Worksheet, and thus a more detailed ALARA review, are structured to ensure that the necessary level of attention is given to each task, job, or project.

7.4.5.2 Conduct of Work in Progress
Radiological work is periodically monitored to ensure proper implementation of controls identified in the ALARA Management Worksheet. Review of work in progress includes monitoring and comparing collective dose accumulation and contamination levels with prejob estimates.

7.4.5.3 Post-job ALARA Review
A formal post-job ALARA review occurs whenever job-specific trigger levels (HSRCM-1, Article 352) are reached. This review compares actual doses with prejob estimates, evaluates the effectiveness of ALARA controls, documents lessons learned, and provides recommendations for similar activities.

7.4.5.4 Internal ALARA Program Reviews and Work Practice Assessments
As mandated by 10 CFR 835 102, “Internal Audits,” the SNF Project conducts an internal audit of its ALARA program at an appropriate frequency and at least once every three years. The internal reviews examine details of radiological work, including items such as work control preparation, prejob briefings, conduct of work, and applied ALARA techniques, post-job briefings, and follow-up discussions.

7.4.5.5 ALARA Records
10 CFR 835 704(b), “Administrative Records,” mandates maintaining ALARA documentation. The SNF Project facilities maintain documentation demonstrating the effectiveness of the ALARA program as described in HSRCM-1, Article 742.
7.5 RADIOLOGICAL PROTECTION TRAINING

Radiological protection training requirements are mandated by 10 CFR 835, Sections 901, 902, and 903. Training requirements are implemented by HSRCM-1, Articles 611 to 657. The general criteria for radiological protection training are detailed below.

7.5.1 General Employee Training

All general employees are trained in radiation safety before receiving occupational exposure during access to controlled areas within the SNF Project facilities. The knowledge of radiation safety possessed by general employees is verified by examination. Retraining is provided when there is a significant change to radiation protection policies and procedures that affects general employees and at intervals not to exceed two years. Allowance may be made for previous DOE training on generic radiation safety topics (i.e., those not specific to a site or facility) provided the training was received at another DOE site or facility within the past two years. Documentation of the previous training shall clearly identify the individual's name, date of training, topics covered, and name of the certifying individual.

7.5.2 Radiological Worker Training

The SNF Project radiological worker training and retraining programs familiarize workers with the fundamentals of radiation protection and the ALARA process and are conducted at intervals not to exceed two years. 10 CFR 835 902 requires a training and retraining program for radiological workers and requires retraining on at least a two-year basis. Radiological worker training not specific to a given site or facility may be waived provided that this training has been received at another DOE site or facility within the past two years and provided that proof-of-training in the form of a certification document containing the individual's name, date of training and specific topics covered and it is certified by an appropriate official. Training includes both classroom and applied training. The knowledge of radiation safety possessed by radiological workers is verified by examination before an unsupervised assignment. The level of training is commensurate with each worker's assignment.

7.5.3 Radiological Control Technician Training

Training and retraining programs for radiological control technicians to familiarize them with the fundamentals of radiation protection and the proper procedures for maintaining exposures ALARA are established and conducted at intervals not to exceed two years. Training includes classroom and applied training. The training either precedes performance of tasks assigned to radiological control technicians or is concurrent with such task assignments if the individual is accompanied by and under the supervision of a trained individual.
The required level of knowledge of radiation safety possessed by radiological control technicians is verified to meet requirements by examination, including on-the-job evaluations before any unsupervised work assignment. The training program includes procedures specific to the individual SNF Project facilities. The level of training is commensurate with the technician's assignment. Records of training are monitored and checked to assure only qualified technicians perform assigned duties. Allowance may be made for previous DOE training on generic radiation safety topics (i.e., those not specific to a site or facility), provided the training was received at another DOE site or facility within the past two years. Documentation of the previous training shall clearly identify the individual's name, date of training, topics covered, and name of the certifying individual.

7.6 RADIATION EXPOSURE CONTROL

The radiation protection program is designed to protect workers from hazards associated with:

- External exposure to radiation
- Spread of radioactive contamination
- Inhalation or ingestion of radioactive materials

The primary engineered mechanisms for ensuring exposure control are shielding from direct radiation and ventilation control of airborne radioactive material. Administrative and monitoring controls for ensuring exposure control include radiological access control, workplace monitoring, and automated radiation area monitors equipped with local alarm annunciators.

HNF-SD-SNF-DB-003 requires that the CVDF and CSB incorporate, as applicable, the following NRC requirements:

- Apply the radiological exposure criteria of Title 10, Code of Federal Regulations, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Section 72 104 "Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS (10 CFR 72 104), to the design and safety analyses.

- Apply the hourly dose limit of Title 10 Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation," Section 20 1301 "Dose Limits for Individual Members of the Public" (10 CFR 20 1301) to the design and safety analysis.

- Incorporate control devices for access to high-radiation areas that conform to the requirements of 10 CFR 20, Section 20 1601, 'Control of Access to High Radiation Areas.'
The 10 CFR 20 1301 hourly dose limit of 0.002 rem (2.0 x 10^{-5} Sv) to the public has been incorporated into the CSB and CVDF designs and safety analyses. This hourly dose limit is assumed to be from external sources for any unrestricted area during normal operations and anticipated occurrences. The 10 CFR 20 1601 requirement for access control devices for high-radiation areas (≥ 0.1 rem/hr at 30 cm) has been applied to the design of the CVDF as discussed in Annex B. The radiological exposure annual dose criteria of 10 CFR 72 104 also have been incorporated into the designs. These criteria apply to design measures to protect any offsite public individual during normal operations and anticipated occurrences both from direct radiation and planned discharges of radioactive materials. These annual dose equivalent criteria are 25 mrem (0.25 mSv) to the whole body, 75 mrem (0.75 mSv) to the thyroid, and 25 mrem (0.25 mSv) to any other critical organ. (Note that existing DOE requirements are 25 mrem [0.25 mSv] to the whole body and 75 mrem [0.75 mSv] to any other critical organ.) The calculated dose to the public from radioactive material releases during normal operation is provided in the radioactive emissions notice of construction for each facility. The dose to the public from direct shine is exceedingly low owing to the low radiation levels at the facilities during operation. The totals of these values are much lower than the 10 CFR 72 104 limits. Refer to Chapter 7.0 of the facility annexes for further description of the dose levels to the public.

7 6 1 Occupational Dose Limits and Administrative Control Levels

Dose limits are derived from 10 CFR 835, Sections 202, 206, 207, and 208. To keep exposures ALARA and to minimize the possibility of exceeding legal personnel exposure limits, the SNF Project has adopted administrative control levels for those exposures that may be received by the workers and visitors. The SNF Project administrative control levels for individuals shown in Table 7-2, are substantially below the regulatory limits shown in Table 7-3 and are multitiered, with increasing levels of authority required to approve higher administrative control levels.

7 6 2 Radiological Practices

Engineered controls are the primary methods used to prevent unnecessary exposure of personnel to radiological hazards. Engineered controls, such as bulk shielding, shielding plugs, canister caps, and shielded work stations, are used to reduce exposure to penetrating radiation from the stored SNF. Exposure to sources of airborne contamination is reduced by sealing potential airborne sources from the atmosphere. Building design, permanent shielding, and confinement systems are described in Chapters 2.0 of the facility FSAR Annexes. Safety and safety-related structures, systems, and components are described in Chapters 4.0 of the facility FSAR Annexes. Technical safety requirements relating to the facility's safety-related structures systems, and components are described in Chapters 5.0 of the facility FSAR Annexes.
Table 7-2  Radiation Worker Administrative Control Levels
(Maximum Annual Dose Equivalent)

<table>
<thead>
<tr>
<th>Total effective dose equivalent (mrem)</th>
<th>Skin or extremity (mrem)*</th>
<th>Lens of eye (mrem)*</th>
<th>Any organ (mrem)*</th>
<th>Approval required to exceed this level (approvals are sequential)</th>
</tr>
</thead>
<tbody>
<tr>
<td>500</td>
<td>15 000</td>
<td>4,500</td>
<td>15,000</td>
<td>Level 3 line manager and radiological control manager</td>
</tr>
<tr>
<td>1,000</td>
<td>22,500</td>
<td>6,750</td>
<td>22,500</td>
<td>Level 2 line manager and radiological control manager</td>
</tr>
<tr>
<td>1,500</td>
<td>30 000</td>
<td>9,000</td>
<td>30,000</td>
<td>Contractor senior Hanford Site executive</td>
</tr>
<tr>
<td>2 000</td>
<td>--</td>
<td>--</td>
<td>--</td>
<td>DOE Program Secretarial Officer</td>
</tr>
</tbody>
</table>

Age x 1 000 = Lifetime cumulative dose equivalent

Level 1 line manager and radiological control manager

Note: Levels are derived from Article 211 of HSRCM 1 Hanford Site Radiological Control Manual
*The values are based on the nonstochastic limit and are calculated as committed doses

DOE = U.S. Department of Energy

Table 7-3  Occupational Dose Limits

<table>
<thead>
<tr>
<th>Type of exposure</th>
<th>Annual limit (rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total effective dose equivalent (internal + external)</td>
<td>5</td>
</tr>
<tr>
<td>Lens of the eye</td>
<td>15</td>
</tr>
<tr>
<td>Extremity (hands and arms below the elbow, feet and legs below the knees)</td>
<td>50</td>
</tr>
<tr>
<td>Any organ or tissue (other than the lens of the eye) and skin</td>
<td>50</td>
</tr>
<tr>
<td>Embryo/fetus</td>
<td>0.5 per gestation period</td>
</tr>
<tr>
<td>Minors whole body (internal + external)</td>
<td>0.1</td>
</tr>
<tr>
<td>Visitors and public (internal + external)</td>
<td>0.1</td>
</tr>
</tbody>
</table>

Note: Levels are from Title 10 Code of Federal Regulations Part 835 Occupational Radiation Protection (10 CFR 835) Refer to 10 CFR 835 for additional detailed information that applies to these limits

General methods for controlling exposure include precautions for conducting radiological tasks and personnel protective equipment. Protocols are established for designating and posting radiological areas and for labeling radioactive material. Entry and exit controls are established and radiological work permits (RWPs) are used to control radiological work. For high-risk radiological work, stay times may be specified and incorporated during radiological work planning processes.
7.6.2.1 Precautions for Conducting Radiological Tasks

Work scopes involving radioactive materials are governed by facility procedures that are approved by facility management. Radiological procedures undergo a review by the radiological control organization to help ensure that all actions necessary to maintain exposure ALARA are incorporated into the procedural steps.

7.6.2.2 Personnel Protective Equipment

Personnel protective equipment includes protective clothing and/or respiratory protection. The amount and type of protective equipment used for a specific task are commensurate with the radiological hazards associated with the task. Specific requirements for protective equipment are identified on the applicable RWPs.

7.6.2.3 Permanent Shielding

Facility-specific permanent shielding is addressed in the facility FSAR Annexes.

7.6.2.4 Multi-Canister Overpack and Transportation Cask Shielding

The MCO transportation cask is designed to minimize dose rates to facility personnel during transfer and processing of the MCO. During processing at the CVDF, the cask lid is removed and the shield plug minimizes the dose to facility personnel during connection and disconnection of the process lines. The SNF Project CSB ALARA Analysis 09 (Transmittal FDP-788) reports that the maximum anticipated personnel dose from the MCOs will be less than 1 rem per year, which meets the 5 rem per year limit required by Section 11.1 of the SNF Project S/RID (HNF-SD-SNF-RD-001) and in 10 CFR 835.202.

An MCO shield plug dose rate analysis has been performed (HNF-SD-SNF-CN-026) to estimate neutron and gamma doses near the MCO. The final ALARA analyses provided with the final facility designs will define whether new shielding or tools are necessary to protect facility personnel. Neutron monitoring will be conducted during initial operations to characterize neutron dose rates. Based on the review of the initial dosimetry and neutron monitoring results, monitoring of the neutron doses may be discontinued. See HNF-SD-SNF-SARR-005 for a more complete description of cask-MCO shielding. Dosimetry records for personnel working near the uncovered MCO will be maintained for beta-gamma and neutron exposures unless neutron dosimetry is discontinued.

7.6.2.5 Radiological Posting and Labeling

Radiological areas are posted in accordance with 10 CFR 835 and S/RID Section 11.1 (HNF-SD-SNF-RD-001) criteria that are summarized in Tables 7-4 and 7-5. Specifications for the color, size, and dimensions of the signs are defined in HSRCM-1.

Posting radioactive material areas is performed in accordance with Article 236 of HSRCM-1. Radiological buffer areas are established within controlled areas to provide secondary boundaries to minimize the spread of contamination and to limit doses to general employees who have not been trained as radiological workers. It is not required that radiological buffer areas be established around inactive or secured contamination areas. The need for radiological buffer areas in conjunction with radioactive material areas is evaluated on a case-by-case basis by the radiological control organization.
Table 7-4  Criteria for Posting Radiation Areas

<table>
<thead>
<tr>
<th>Area</th>
<th>Dose rate criteria</th>
<th>Posting</th>
</tr>
</thead>
<tbody>
<tr>
<td>Radiation area</td>
<td>&gt;0 005 rem/h and ≤0 1 rem/h at 30 cm</td>
<td>&quot;CAUTION, RADIATION AREA&quot;</td>
</tr>
<tr>
<td></td>
<td></td>
<td>&quot;Personnel Dosimeter Required for Entry&quot;</td>
</tr>
<tr>
<td>High radiation area</td>
<td>&gt;0 1 rem/h at 30 cm and ≤500 rad/h at 100 cm</td>
<td>&quot;DANGER, HIGH RADIATION AREA&quot;</td>
</tr>
<tr>
<td></td>
<td></td>
<td>&quot;Personnel Dosimeter, Supplemental Dosimeter and RWP Required for Entry&quot;</td>
</tr>
<tr>
<td>Very high radiation area</td>
<td>&gt;500 rad/h at 100 cm</td>
<td>&quot;GRAVE DANGER, VERY HIGH RADIATION AREA&quot;</td>
</tr>
<tr>
<td></td>
<td></td>
<td>&quot;SPECIAL CONTROLS REQUIRED FOR ENTRY&quot;</td>
</tr>
<tr>
<td>Hot spot</td>
<td>5 times general area dose rate and &gt;0 1 rem/h</td>
<td>&quot;CAUTION, HOT SPOT&quot;</td>
</tr>
</tbody>
</table>

Note: Refer to Title 10 Code of Federal Regulations Part 835 Occupational Radiation Protection (10 CFR 835) and HSRCM 1 Hanford Site Radiological Control Manual for additional information that may apply.

RWP = radiological work permit

Table 7-5  Criteria for Posting Contamination, High Contamination, and Airborne Radioactivity Areas (2 sheets)

<table>
<thead>
<tr>
<th>Area</th>
<th>Criteria</th>
<th>Posting</th>
</tr>
</thead>
<tbody>
<tr>
<td>Contamination</td>
<td>Contamination levels (dpm/100 cm²) &gt;1 times but ≤100 times HSRCM-1⁸ Table 2-2 values</td>
<td>&quot;CAUTION, CONTAMINATION AREA&quot;</td>
</tr>
<tr>
<td>High contamination</td>
<td>Contamination levels (dpm/100 cm²) &gt;100 times HSRCM-1⁸ Table 2-2 values</td>
<td>&quot;DANGER, HIGH CONTAMINATION AREA&quot;</td>
</tr>
<tr>
<td></td>
<td></td>
<td>&quot;RWP Required for Entry&quot;</td>
</tr>
<tr>
<td>Fixed contamination</td>
<td>Removable contamination levels &lt; HSRCM-1⁸ Table 2-2 values, and total contamination levels &gt; Table 2-2 total values</td>
<td>&quot;CAUTION FIXED CONTAMINATION&quot;</td>
</tr>
<tr>
<td>Soil contamination</td>
<td>Contaminated soil not releasable in accordance with DOE Order 5400 5⁵</td>
<td>&quot;CAUTION, SOIL CONTAMINATION AREA&quot;</td>
</tr>
</tbody>
</table>
Table 7-5  Criteria for Posting Contamination, High Contamination, and Airborne Radioactivity Areas  (2 sheets)

<table>
<thead>
<tr>
<th>Area</th>
<th>Criteria</th>
<th>Posting</th>
</tr>
</thead>
</table>
| Airborne radioactivity| Concentrations (μCi/cm³) >10% of any DAC value listed in 10 CFR 835<sup>5</sup> | "CAUTION, AIRBORNE RADIOACTIVITY AREA"  
"RWP Required for Entry" |

Note Refer to the referenced documents for additional information that may apply
"DOE Order 5400 5  Radiation Protection of the Public and the Environment  U S Department of Energy Washington, D C
10 CFR 835  Occupational Radiation Protection  Code of Federal Regulations  as amended

DAC = derived air concentration
dpm = disintegrations per minute
RWP = radiological work permit

7.6.2.6 Radiological Access Controls  Radiological access controls are administered in accordance with the requirements of Section 11.1 of the SNF Project S/RID (HNF-SD-SNF-RD-001) and of 10 CFR 835, as discussed below. Access control is maintained for each radiological area to a degree commensurate with the existing and potential radiological hazards within the area. One or more of the following methods is used to ensure access control:

- Signs and barricades
- Control devices on entrances
- Conspicuous visual and/or audible alarms
- Locked entrance ways
- Administrative controls

The FSAR annexes provide additional detail on the specific access control methods to be used at the individual facilities.

Administrative procedures are written to demonstrate compliance with 10 CFR 835 and include actions essential to ensure the effectiveness and operability of barricades, devices, alarms and locks. Authorizations are required to perform specific work within radiological areas and are affiliated with procedures that include specific radiation protection measures.

Access control authorizations are administered in conjunction with computerized database systems that enable supervisors and individuals to assess their to-date exposure and exposure levels. This information is provided as a tool to be used in determining and administering stay times in radiological areas for the purpose of using exposure effectively and in accordance with ALARA principles. Entryways to high radiation areas (>0 1 rem/h at 30 cm) will be locked or...
guarded at all times. No controls are installed that would prevent rapid evacuation of personnel under emergency conditions [10 CFR 835 501(e) and 10 CFR 835 502(c)].

7 6 2 7 Radiological Work Permits The RWP is the administrative radiological control tool for intended work activities at SNF Project facilities. The RWP is used as specified in HSRCM-1, Article 321, to govern entries into radiological areas and to control all radiological work. RWPs are issued by the radiological control organization and are used by radiological workers. The permits define the protective clothing requirements, maximum allowable contamination and radiation levels, respiratory protection requirements, dosimetry requirements, radiological worker training requirements, and special precautions relative to job-specific activities. RWPs generally are prepared in conjunction with work planning activities and/or procedure development efforts and are part of the work package. RWPs are updated if radiological conditions change to the extent that protective requirements need modification. They also provide a mechanism to relate worker exposure to specific work activities.

7 6 3 Dosimetry

External dosimetry is provided to and used by all radiological workers who, under typical conditions, are likely to receive exposures that trigger one or more of the criteria identified in Table 7-6. These criteria comply with the requirements of S/RID Section 11 1 (HNF-SD-SNF-RD-001) and 10 CFR 835 402 "Individual Monitoring."

<table>
<thead>
<tr>
<th>Exposure type</th>
<th>Exposure value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Whole body effective dose equivalent</td>
<td>0.1 rem/yr</td>
</tr>
<tr>
<td>Shallow dose to the skin or any extremity</td>
<td>5 rem/yr</td>
</tr>
<tr>
<td>Lens of the eye dose equivalent</td>
<td>1.5 rem/yr</td>
</tr>
<tr>
<td>Deep dose equivalent from external exposure to any organ or tissue other than the lens of the eye</td>
<td>5 rem</td>
</tr>
<tr>
<td>Embryo/fetus</td>
<td>0.05 rem/gestation</td>
</tr>
<tr>
<td>Minors/members of the public</td>
<td>0.05 rem/yr</td>
</tr>
<tr>
<td>Individuals entering high or very high radiation areas</td>
<td>All entries require dosimetry</td>
</tr>
</tbody>
</table>

Personnel working at SNF Project facilities are not expected to receive occupation radiological exposure from internally deposited radioactive material. To minimize airborne contamination, which could lead to internal deposition, the facility designs incorporate area...
ventilation systems and/or spot ventilation systems in key locations. Chapters 20 and 70 of the facility FSAR Annexes contain facility-specific ventilation systems information. Radiological workers who meet trigger levels (identified in Table 7-7) for potential internal deposition are included in a bioassay program consistent with the requirements in 10 CFR 835 and HSRCM-1.

### Table 7-7 General Trigger Levels that Drive Requirements for Internal Dosimetry

<table>
<thead>
<tr>
<th>Exposure type</th>
<th>Exposure value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Committed effective dose equivalent (whole body) from radionuclide intakes</td>
<td>0.1 rem/yr</td>
</tr>
<tr>
<td>Committed dose equivalent to any organ or tissue from radionuclide intakes</td>
<td>5 rem/yr</td>
</tr>
<tr>
<td>Dose equivalent to the embryo/fetus from radionuclide intakes</td>
<td>0.05 rem/gestation</td>
</tr>
<tr>
<td>Committed effective dose equivalent to minors/members of the public</td>
<td>0.05 rem/yr</td>
</tr>
</tbody>
</table>

The radiological control organization issues supplemental dosimetry on a job-specific basis in accordance with the applicable RWPs. Hanford Site standard dosimetry exchanges are coordinated with the central dosimetry organization on a schedule that is commensurate with the anticipated annual exposure for each radiological worker (annually as a minimum). Personnel working near an MCO that has the cask lid removed will have their beta-gamma and neutron exposures recorded. Based on the review of the initial dosimetry results, monitoring of neutron doses may be discontinued.

### 7.6.4 Respiratory Protection

Respiratory protection criteria are described in HSRCM-1. Use of respiratory protection is reduced to the minimum level practicable by implementing engineering controls and work practices to contain radioactivity at the source. The facility designs do not require routine respirator use except at the K Basins. Radiological monitoring during initial MCO processing will validate the design intent relative to respirator use. Respirator use may be required during certain maintenance and overhaul activities at SNF Project facilities not considered to be normal routine operations. The radiological control organization performs reviews of procedures during the planning stages for radiological work. Specific engineering controls are implemented on a job-specific basis and are commensurate with the degree of hazard associated with the anticipated work scope.
Standards and requirements for implementing a respiratory protection program and associated training of personnel are driven by HSRCM-1. Respiratory protection described in this safety analysis report is not the safety analysis basis for preventing inhalation of radionuclides among radiological workers. The engineering controls associated with the design of the SNF Project facilities are designed to prevent the introduction of radioactive airborne contaminants into the work environment. Engineering controls represent the primary basis for ensuring radiological safety of workers.

The Respiratory Protection Program (HNF-PRO-120) applies to nonemergency activities that require respiratory protection for hazards including radiological, chemical, and particulate. The Respiratory Protection Program governs the purchase, selection, issuance and control, use, maintenance, and quality assurance for all respirators used by SNF Project workers. Medical qualification for respirator wearers is identified through the worker job task analysis processes and the occupational medical provider is responsible for the medical evaluation and qualification of respirator wearers. Respiratory protection complies with the requirements of 29 CFR 1910.134 and HSRCM-1.

The SNF Project respiratory protection program complies with the respiratory protection standards published in ANSI Z88.2-1992, *Respiratory Protection*, and includes the following elements:

- A written policy statement
- Guidance on proper selection of equipment, based on the hazard
- Requirements for fitting, use, cleaning, storage, inspection, and maintenance of respiratory equipment
- Use of respiratory protection equipment that has the approval of the National Institute for Occupational Safety and Health

The type of respirator to be used is specified in writing on the applicable job hazard analysis maintenance procedure, operating procedure, or job-specific permit.

### 7.7 Radiological Monitoring

Section 11.1 of the SNF Project S/RID (HNF-SD-SNF-RD-001) 10 CFR 835.401 and 10 CFR 20 define the required program for radiological monitoring and surveys. HSRCM-1 Article 551 specifies that scheduled radiation survey task descriptions provide the field direction to radiological control technicians performing required surveillances. These task descriptions schedule the specific regions of the facility to be surveyed and the types and frequencies of surveys to be performed and clarify the minimum documentation requirements for completed surveillances. Task descriptions are written to satisfy the requirements of HSRCM-1.
7.7.1 Radiation Exposure Surveys

Radiation exposure survey programs are established in accordance with 10 CFR 835 401 and are based on potential radiological conditions, probability of change in conditions, and area occupancy factors.

7.7.2 Contamination Surveys

Contamination survey programs are established in accordance with 10 CFR 835 401 and 10 CFR 835 404. All radiological areas, as defined by HSRCM-1, will be routinely surveyed based on type of radiological area, probability of change in conditions, potential for the spread of contamination, and area occupancy factors.

In addition to these routine contamination survey programs, surveys also are performed:

- During initial entry into known or suspected contamination areas
- Periodically during work involving the potential for radioactive contamination
- At the completion of work involving the potential for radioactive contamination
- As specified by RWPs and scheduled radiation survey task descriptions
- Following a leak or spill of radioactive materials

7.7.3 Monitoring for Radioactive Airborne Contaminants

Criteria for monitoring radioactive airborne contaminants are mandated by Section 111 of the SNF Project S/RID (HNF-SD-SNF-RD-001) and by 10 CFR 835 403. Selection of air monitoring equipment is based on the specific job being monitored. Air monitoring equipment includes portable and fixed air sampling equipment and continuous air monitors.

Air sampling equipment is used in occupied areas where, under normal operating conditions, a person is likely to receive an annual intake of 2% or more of the specified annual limit-of-intake values (40 DAC [derived concentration] hours).

Continuous air monitoring equipment is installed in occupied areas where a person without respiratory protection may be exposed to a concentration of radioactivity in air exceeding 1 DAC or where there is a need to alert potentially exposed workers to unexpected increases in the airborne radioactive material levels.

7.7.4 Monitoring Records

Required surveillances are documented, and the survey records are subject to the records retention criteria of 10 CFR 835 703. Data collected during radiological monitoring activities are recorded according to procedure and made available to appropriate staff for responding to action...
levels, assessing the effectiveness of radiological protection programs, preparing lessons-learned information, and archiving.

7.8 RADIOLOGICAL PROTECTION INSTRUMENTATION

In accordance with 10 CFR 835.401, instruments used for monitoring and contamination control shall be

- Periodically maintained and calibrated on an established frequency of at least once per year
- Appropriate for the types, levels, and energies of the radiation encountered
- Appropriate for existing environmental conditions
- Routinely tested for operability

General criteria for radiological instrumentation are provided by HSRCM-1, Articles 551 through 564.

7.8.1 Selection and Placement Criteria

7.8.1.1 Air Sampling Equipment  In general, air sampling equipment is located in the vicinity of workers to provide an indication of the airborne radioactive material levels to which the workers are exposed. Air monitoring equipment is used in situations where airborne radioactivity levels can fluctuate and early detection of airborne radioactivity could prevent or minimize inhalation of radioactivity by personnel. Air sampling equipment placement criteria are documented in HSRCM-1, Article 555, and the facility technical basis document. The criteria in these documents are based on airflow patterns in the facility, work activity, quantity and type of contamination present or anticipated, and the location of workers.

7.8.1.2 Area Radiation Monitoring  Monitoring of radiation in the workplace is performed using stationary (area) and portable radiation instruments. S/RID Section 11.1 (HNF-SD-SNF-RD-001) require monitoring of individuals and areas to accomplish the following objectives:

- Detect changes in radiological conditions
- Detect gradual buildup of radioactive material in the workplace
- Verify the effectiveness of engineering and process controls in containing radioactive material and reducing radiation exposure
• Document radiological conditions

• Demonstrate compliance to requirements

Area radiation monitors are designed into SNF Project facilities or are added to the facility based on these criteria. The specific location of area radiation monitors is described in the facility FSAR Annexes. Monitoring instruments in SNF Project facilities are equipped to warn radiological workers of adverse radiological conditions so they can evacuate the effected region in time to minimize their exposure. Monitoring with portable instruments is performed utilizing appropriately selected, maintained and calibrated instruments. These instruments are appropriate to the radiations encountered and the environmental conditions and are routinely tested for operability.

7 8 1 3 Automated Personnel Monitors SNF Project facilities S/RID Section 111 (HNF-SD-SNF-RD-001) require that monitoring to detect and prevent the spread of contamination be performed by personnel exiting radiological areas. Instruments and techniques used for this monitoring are required to be effective in controlling contamination. At SNF Project facilities automated personnel contamination monitors are provided for personnel monitoring and contamination control in addition to portable instruments at commonly used egress points. These devices include beta–gamma detection, as appropriate. The exact locations of personnel survey devices are subject to considerations that include background interference, radiological postings, and egress pathways. General criteria for personnel release surveys are defined by HSRCM-1, Article 338. See the facility FSAR Annexes for specific applications of automated personnel monitors.

7 8 1 4 Portable and Laboratory Radiation Monitoring Equipment A variety of portable and laboratory equipment is used at SNF Project facilities. General criteria for radiological instrumentation are addressed by HSRCM-1, Articles 551 to 564.

7 8 2 Types of Detectors

A summary of the different kinds of radiation detection instruments used at SNF Project facilities is presented below:

• Body frisker
• Hand and foot monitors
• Portal monitors
• Area radiation monitors
• Fixed head air samplers
• Beta–gamma continuous air monitors
• Alpha continuous air monitors

This is a general list. Specific application of instrument types is addressed in the facility FSAR Annexes.
7.8.3 Instrument Calibration

Radiation protection instrumentation is calibrated in accordance with Section 11.1 of the SNF Project S/RID (HNF-SD-SNF-RD-001) and 10 CFR 835 401(c). Calibrations are performed at least annually in accordance with established procedures that meet the requirements of ANSI N323, *Radiation Protection Instrumentation Test and Calibration*. All calibrations use National Institute of Standards and Technology-traceable sources. Standard radiation sources are used to routinely check instruments in the field. Administrative controls, procedures, and training ensure the instruments used are appropriate for the type(s), levels, and energies of the radiation(s) encountered and are appropriate for existing environmental conditions.

7.8.4 Quality Assurance for Calibration and Maintenance

The SNF Project quality assurance program is based on Title 10 *Code of Federal Regulations*, Part 830, "Nuclear Safety Management." Section 830.120 "Quality Assurance Requirements" (10 CFR 830.120). Radiological instruments are selected to be appropriate for the field conditions and environment in which they are expected to operate, and they are used to measure only the radiation for which their calibrations are valid.

Calibrations are performed using National Institute of Standards and Technology-traceable standards and procedures developed for each radiological instrument type. Keeping the calibration of instruments current is ensured by affixing a label or tag with the date of calibration and the date recalibration is due. For instruments whose "as found" readings indicate that the instrument may have been used while out of calibration, immediate notification of the out-of-calibration condition is made to the appropriate radiological control organization office. The radiological control organization office reviews any measurements performed with the instrument while it was suspected of being out of calibration.

A program for preventive and corrective maintenance of radiological instruments has been established. The components and procedural recommendations used for preventive and corrective maintenance are at least as stringent as those specified by the instrument's manufacturer. After any preventive or corrective maintenance or any adjustment that voids the previous calibration, radiological instruments undergo calibration before use.

7.9 RADIOLOGICAL PROTECTION RECORD KEEPING

Radiological protection record keeping requirements are mandated by SNF Project facility S/RID Section 11.1 (HNF-SD-SNF-RD-001). Table 7.8 identifies the primary records that are applicable to radiation protection and a general overview of how these records are integrated into the facility record system. Radiological protection records that relate to the operation of the radiological protection program (e.g., RWPs, ALARA reviews, radiological surveys, and radiological problem reports) are maintained by the SNF Project radiological control organization according to SNF Project administrative procedures and quality assurance manuals.

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Table 7-8 Radiological Records

<table>
<thead>
<tr>
<th>Type of record</th>
<th>Administrative program</th>
</tr>
</thead>
<tbody>
<tr>
<td>Radiological policy statements</td>
<td>Facility and PHMS administrative procedures</td>
</tr>
<tr>
<td>Radiological control procedures</td>
<td>Facility and PHMS technical work documents and technical</td>
</tr>
<tr>
<td>doses</td>
<td>procedures</td>
</tr>
<tr>
<td>Individual radiological doses</td>
<td>Dosimetry reporting system</td>
</tr>
<tr>
<td>Internal and external dosimetry policies and procedures</td>
<td>PHMS technical procedures</td>
</tr>
<tr>
<td>Personnel training</td>
<td>Facility training matrix database</td>
</tr>
<tr>
<td>ALARA records</td>
<td>Job control system work packages</td>
</tr>
<tr>
<td>Instrumentation records</td>
<td>Task descriptions  calibration facility reports job control</td>
</tr>
<tr>
<td>Radiological surveys</td>
<td>system work packages</td>
</tr>
<tr>
<td>Area monitoring dosimetry results</td>
<td>Radiological survey reports</td>
</tr>
<tr>
<td>Radiological work permits</td>
<td>Radiological work permit record system</td>
</tr>
<tr>
<td>Performance indicators and assessments</td>
<td>Periodic reports/technical review documents</td>
</tr>
<tr>
<td>Radiological safety analysis and evaluation reports</td>
<td>Supporting documents/technical review documents</td>
</tr>
<tr>
<td>Quality assurance records</td>
<td>Supporting documents/technical review documents</td>
</tr>
<tr>
<td>Radiological incident and occurrence reports</td>
<td>Critique reports occurrence reports unusual occurrence</td>
</tr>
<tr>
<td>reports</td>
<td>reports unreviewed safety question reports</td>
</tr>
<tr>
<td>Radioactive sources</td>
<td>Radioactive source custodian records</td>
</tr>
<tr>
<td>Radiological release surveys</td>
<td>Radiological survey reports</td>
</tr>
<tr>
<td>Reports of lost radioactive material</td>
<td>Radiological survey reports occurrence reports supporting</td>
</tr>
<tr>
<td></td>
<td>documents</td>
</tr>
</tbody>
</table>

ALARA = as low as reasonably achievable  
PHMS = Project Hanford Management System

7.10 OCCUPATIONAL RADIATION EXPOSURES

The facility FSAR Annexes contain discussions on the estimated occupational radiation exposures

7.11 REFERENCES


DOE Order 5400 5 Radiation Protection of the Public and the Environment U S Department of Energy Washington, D C

DOE Order 5480 21 Unreviewed Safety Questions U S Department of Energy Washington D C

FDP-788 1998, SNF Canister Storage Building ALARA Analysis 09, Fluor Daniel Incorporated, Richland Washington

HNF-PRO-120 Respiratory Protection Program Fluor Daniel Hanford Incorporated Richland Washington

HNF-PRO-1618 ALARA Policy and Management Commitment Fluor Daniel Hanford Incorporated Richland Washington

HNF-PRO-1619 ALARA Organization and Responsibilities, Fluor Daniel Hanford Incorporated Richland, Washington

HNF-PRO-1620 ALARA Program Scope Fluor Daniel Hanford Incorporated Richland Washington

HNF-PRO-1621 ALARA Decision-Making Methods, Fluor Daniel Hanford Incorporated Richland Washington

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HNF-PRO-1629 Administrative Control Levels Fluor Daniel Hanford Incorporated, Richland, Washington

HNF-PRO-1631, ALARA Training, Fluor Daniel Hanford Incorporated, Richland Washington

HNF-PRO-1633, ALARA Program Records, Fluor Daniel Hanford, Incorporated Richland Washington

HNF-SD-SNF-CN-026 1997 MCO Shield Plug Dose Rate Analysis, Rev 0, Fluor Daniel Hanford Incorporated Richland, Washington


NRC Regulatory Guide 8 8 1978 Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable Rev 3 U S Nuclear Regulatory Commission Washington D C
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CHAPTER 80

HAZARDOUS MATERIAL PROTECTION
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# LIST OF TERMS

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<thead>
<tr>
<th>Abbreviation</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>ACGIH</td>
<td>American Conference of Governmental Industrial Hygienists</td>
</tr>
<tr>
<td>ALARA</td>
<td>as low as reasonably achievable</td>
</tr>
<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
</tr>
<tr>
<td>EJTA</td>
<td>employee job task analysis</td>
</tr>
<tr>
<td>HAZWOPER</td>
<td>hazardous waste operation</td>
</tr>
<tr>
<td>HGET</td>
<td>Hanford General Employee Training</td>
</tr>
<tr>
<td>JHA</td>
<td>job hazard analysis</td>
</tr>
<tr>
<td>JSA</td>
<td>job safety analysis</td>
</tr>
<tr>
<td>MSDS</td>
<td>material safety data sheet</td>
</tr>
<tr>
<td>OSHA</td>
<td>Occupational Safety and Health Administration</td>
</tr>
<tr>
<td>PEL</td>
<td>permissible exposure limit</td>
</tr>
<tr>
<td>PPE</td>
<td>personal protective equipment</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
</tr>
<tr>
<td>S/RID</td>
<td>standards/requirements identification document</td>
</tr>
<tr>
<td>TLV</td>
<td>threshold limit value</td>
</tr>
<tr>
<td>VPP</td>
<td>Voluntary Protection Program</td>
</tr>
</tbody>
</table>
8 0 HA ZA RD OUS MATERI AL PRO TE C T I ON

8 1 I NTRO DUC T I ON

This chapter addresses the major provisions of the occupational safety and health program as it applies to hazardous material protection for the Spent Nuclear Fuel (SNF) Project. Hazardous material, as defined in U.S. Department of Energy (DOE) Order 5480 23, *Nuclear Safety Analysis Reports*, is "any solid, liquid, or gaseous material that is toxic, explosive, flammable, corrosive, or otherwise physically or biologically threatening to health. Oil is excluded from this definition." This chapter contains sufficient information to demonstrate compliance with applicable requirements for control of personnel exposures to hazardous materials.

The SNF Project is cognizant of the requirements applicable to a hazardous material protection program. An effective SNF Project occupational safety and health program will be used to control onsite worker exposures to hazardous materials and will provide the protection necessary to control exposures to the general public.

8 2 R EQ UIRE ME NTS

The requirements that form the basis for the hazardous material protection program are found in the SNF Project standards/requirements identification document (S/RID) (HNF-SD-SNF-RD-001). The specific requirements applicable to this chapter include:

- DOE Order 5480 4 *Environmental Protection Safety and Health Protection Standards*
- DOE Order 5480 8A, *Contract Occupational Medical Program*
- DOE Order 5480 9A, *Construction Project Safety and Health Management*
- DOE Order 5480 10, *Contractor Industrial Hygiene Program*
- DOE Order 5483 1A, *Occupational Safety and Health Program for DOE Contractor Employees at Government Owned Contractor-Operated Facilities*
8.3 HAZARDOUS MATERIAL PROTECTION PROGRAM AND ORGANIZATION

The SNF Project has an established, visible, and comprehensive occupational safety and health program that implements the occupational health and safety requirements and guidelines of DOE Order 5480.10 as well as other relevant DOE and DOE Richland Operations Office directives described in Section 8.2. The goal of the SNF Project occupational safety and health program is to control worker exposure to work hazards within the limits prescribed by DOE and by professional occupational safety and health practices and principles. The primary objective is to provide a workplace free from or protected against hazards likely to cause injury, illness, impaired health and well-being, or significant discomfort or inefficiency among workers or the community.

The occupational safety and health program uses the Occupational Safety and Health Administration (OSHA) permissible exposure limits (PELs) and the American Conference of Governmental Industrial Hygienists (ACGIH) threshold limit values (TLVs) to provide guidance on controlling exposures and protecting the worker. Where a lower published exposure limit exists, the SNF Project will use that limit, according to the hierarchy published in DOE Order 5480.10. Workers will be knowledgeable of such limits and standards and of SNF Project policies and procedures through applicable training (see Section 8.5). All SNF Project employees are instructed to observe DOE-prescribed OSHA standards applicable to this work and report promptly any conditions that may lead to a violation of these standards in accordance with DOE Order 5483.1A. The Hanford Fire Department acts as the incident command agency for emergency incidents involving hazardous materials.

The SNF Project implements the hazardous material protection program based on the requirements of HNF-MP-003, *Integrated Environment Safety and Health Management System Plan*. More detailed guidance is provided in SNF Project administrative procedures.

The SNF Project's occupational safety and health organization is responsible for administering the occupational safety and health program. The SNF Project Occupational Safety and Health Manager reports directly to the SNF Project Vice President/Project Director's office, who has ultimate responsibility for the hazardous material program implementation. This includes responsibility for establishing the occupational safety and health program in accordance with DOE requirements and other applicable requirements documents and providing appropriate resources to support the occupational safety and health programs.

The SNF Project implementing procedures reflect the management commitment and worker involvement facets of the DOE Voluntary Protection Program (VPP) provided in DOE/EH-0433, *U.S. Department of Energy Voluntary Protection Program* and in HNF-MP-003. Safety training of SNF Project management and workers places an emphasis on the elements of the VPP, and workers are encouraged to actively participate in VPP initiatives.

The administrative activities and responsibilities of the support organization are summarized in the following sections.
8.3.1 Organizational Structure

The effectiveness of the hazardous material program depends on the formal definition and communication of organizational structure, responsibilities, and authorities through the application of organization charts and job descriptions. The organizational structure of the SNF Project, including staffing levels and qualifications, positions of authority, responsibilities, and interfaces with other safety organizations and facility operations, is discussed in Chapter 17. Basic rights and responsibilities of SNF Project personnel are also described in HNF-PRO-074 Safety Responsibilities. Therefore, this section summarizes the responsibilities of the organizations with regard to the SNF Project occupational safety and health program in accordance with requirements of Section 19.0 of the SNF Project S/RID (HNF-SD-SNF-RD-001). Management self-assessments will be used by management to verify that the requirements are being met and/or implemented and will be used to measure performance objectives.

8.3.1.1 Spent Nuclear Fuel Project Operations Managers. The SNF Project Operations Managers bear full responsibility and accountability for providing workers a safe and healthy working environment. The SNF Project Operations Managers report to the SNF Project Vice President/Project Director's office and have the following hazardous material protection program responsibilities:

- Establish and maintain safe conditions in the workplace.
- Identify and monitor, on a regular basis, the hazards and associated risks that may exist at the facility.
- Account for modifications and changes that may occur to the equipment and/or processes.
- Provide all workers with the hazard information and training necessary to perform their jobs in a safe and healthy manner.
- Ensure that workers have all necessary medical clearances for their jobs and enforce all medical work restrictions.
- Ensure that workers assigned to tasks involving hazardous materials have the appropriate training required for the job.
- Promptly inform the occupational safety and health organization of any operation or condition that appears to present a hazard to workers.
- Provide personal protective equipment as specified by the occupational safety and health organization and ensure that workers wear this equipment and undergo training to use it properly.
• Provide funding for occupational safety and health assessments and hazard control

• Take appropriate disciplinary action for violations of health and safety rules

• Promptly respond to formal occupational safety and health recommendations for needed control measures

• Perform accident investigations

• Incorporate provisions for protection of the public, workers and environment into the conduct of operations program

• Identify the procedures and controls that must be followed to guard against the hazards

• Conduct operations consistent with established health and safety procedures

• Develop and implement a chemical management program

• Provide a hazard communication program

• Perform SNF Project emergency planning and response actions as defined in Chapter 15

• Include industrial hygiene staff in the design review process whenever new construction or remodeling of an existing process is planned

• Request industry hygiene staff to evaluate effectiveness of proposed environment control equipment and approve procedure for use

8.3.1.2 Spent Nuclear Fuel Project Occupational Safety and Health Manager The SNF Project Occupational Safety and Health Manager has the following responsibilities

• Develop and administer programs to control and minimize health and safety hazards and to promote continuous improvement in safety and health performance

• Provide safety and health expertise, assistance and support for SNF Project operations and activities

• Maintain safety and health records and administrative controls and coordinate with Hanford Site service organizations as necessary

• Provide and manage resources to design and implement occupational safety and health programs
• Ensure that applicable requirements are integrated into the occupational safety and health program and the procedures and standards of other applicable organizations

• Procure and maintain hazardous material monitoring instrumentation

• Review and evaluate Hanford Site and SNF Project facility-specific training programs to ensure adequacy of occupational safety and health components with the occupational safety and health organization participating, as necessary, as subject matter experts in training program development and delivery

• Support professional development of the occupational safety and health organization staff by sending the staff to onsite and offsite technical training and encourage individual initiative by supporting efforts to obtain necessary certification

8 3 1 3 Spent Nuclear Fuel Project Engineering Manager (Chief Engineer) The SNF Project Chief Engineer/Director has the following responsibilities

• Provide assistance and support in hazard assessments, hazard posting, and container labeling

• Provide technical support to the maintenance of the facility's hazardous chemical product lists and interface with work control and operating facilities as requested

• Use established approved engineering procedures to minimize exposures to harmful environmental factors or stresses

• Notify the occupational safety and health organization whenever new operations or processes are to be introduced so that potential hazards may be evaluated in the planning stage

• Obtain occupational safety and health program approval for new installations and equipment before its initial use

8 3 1 4 Spent Nuclear Fuel Project Technical Training SNF Project technical training is responsible for establishing, conducting and administering the training program for SNF Project facility managers to ensure that all workers meet the hazardous material protection training requirements of their assigned tasks

8 3 1 5 Spent Nuclear Fuel Project Workers Each worker is responsible for recognizing and promptly notifying management of events and conditions that could have adverse safety or environmental implications. Workers are also responsible for obeying all oral and written hazardous material control instructions and procedures conscientiously applying all health and safety training in daily activities, and properly using all prescribed personal protective equipment. In addition, SNF Project workers have the following specific responsibilities...
Observe the Master Safety Rules
Perform work in accordance with the specified work instructions, work procedures, and standard operating procedures
Use all prescribed personal protective equipment
Actively support programs designed to protect workers
Stop work to prevent or control hazards considered to be an immediate threat
Work defensively and be vigilant that coworkers do not put themselves at risk
Submit ideas for safety and health improvements to management
Report injuries and medically imposed work restrictions to the immediate manager in a timely manner

**8 3 1 6 Occupational Medical Services Contractor** An external organization is contracted by DOE to provide occupational medical services. As described in Section 8 6 3 and Chapter 17 0, the SNF Project relies on this occupational medical support organization for the following services

- Performing preplacement health appraisals, periodic health examinations, and health education programs
- Providing appropriate physical examinations for workers who work with, or are exposed to, hazardous agents or materials
- Maintaining records of occupational illnesses in accordance with the requirements of Title 29, *Code of Federal Regulations* Part 1904, Recording and Reporting Occupational Injuries and Illnesses (29 CFR 1904)
- Setting medical restrictions for workers whenever warranted by examination findings and ensuring that the restricted worker's management is notified of the restrictions
- Ensuring that the bioassay and medical monitoring programs are responsive to the requirements of 29 CFR 1910 120 and 29 CFR 1910 Subpart Z
- Consulting with the industrial hygiene staff on the need for job-related medical examinations or bioassays
- Assisting in performance of workplace hazard evaluations as requested
8317  Spent Nuclear Fuel Project Industrial Hygiene  As required by Section 19.2 of the SNF Project S/RID (HNF-SD-SNF-RD-001), the SNF Project industrial hygiene staff is professionally qualified, adequately staffed and has sufficient time and authority to design and implement the industrial hygiene program. Industrial hygiene has the following responsibilities:

- Alerting the industrial hygiene staff to all suspected occupational illnesses to facilitate early evaluation and correction of problems.
- Maintaining medical records in accordance with the applicable OSHA and DOE requirements.
- Providing first aid services including diagnosis and treatment of non-life threatening medical events.
- Providing technical assistance in areas such as chemical exposure assessment and collection of personal monitoring data to ensure necessary calculations to determine exposure and recommend appropriate hazard controls.
- Notify workers on results of personal exposure monitoring.
- Recommend workers to applicable managers to be included in medical surveillance programs.
- Exercise the traditional tenets of industrial hygiene in anticipating, recognizing, evaluating, and controlling hazards in the work place.
- Assist in the preparation, review, and approval of job hazard analyses (JHAs), job safety analyses (JSAs), automated JHAs, automated employee job task analyses (EJTAs), work plans and packages, preventive maintenance actions, plant operating procedures, and other technical and operations-related documents where industrial hygiene is a potential issue.
- Review the material safety data sheets (MSDSs) for toxic materials and new safety products or related equipment materials purchases.
- Communicate hazard information to workers and their managers so the workers and managers can perform their jobs safely.
- Provide for the collection and analysis of necessary samples and ensure that instrumentation is properly calibrated, used, and maintained.
- Collect and maintain accurate and precise information on occupational health hazards and preserve and maintain the information in a centralized system.
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- Develop occupational safety and health program criteria and the standards necessary to ensure proper identification, evaluation, and control of workplace health hazards.
- Conduct periodic audits and appraisals to monitor compliance with the requirements of the occupational safety and health procedures and to evaluate the overall effectiveness of the program.
- Approve hazardous material requisitions and sign the purchase requisition to indicate approval before the material is purchased.
- Verify conformance to chemical management requirements such as safe storage, usage, tracking, and substitutions of less hazardous chemicals.
- Interface with DOE Richland Operations Office to assess and resolve concerns.

8 3 1 8 Spent Nuclear Fuel Project Chemical Management Point of Contact: The chemical management point of contact has the following responsibilities:
- Provide technical support and expertise in assisting SNF Project and activities.
- Provide pre-acquisition reviews of material and service requests to determine if less hazardous substitutes are available or if item is already in stock and current inventory can be used.
- Provide information with regards to chemical inventories, chemical usage, and chemical release information for required reporting of hazardous chemicals.
- Communicate hazard information to workers and their managers so the workers and managers can perform their jobs safely.
- Verify conformance to chemical management requirements such as safe acquisition, transportation, storage, usage, tracking, and substitutions of less hazardous chemicals.
- Act as point of contact for all chemical management issues involving other Project Hanford Management Contract Major Subcontractors, subcontractors, and all other non-SNF Project personnel conducting work at SNF Project facilities.
- Maintain the facility’s hazardous chemical product lists and provide lists to work control and project facilities.

8 3 1 9 Contracts: The SNF Project Contracts organization has the following occupational safety and health responsibilities:
- Control the purchase and distribution of potentially hazardous materials.
Obtain assistance from the occupational safety and health department when hazardous materials are requested.

Ensure that MSDSs are received with each shipment from the chemical manufacturer or supplier and submitted to the MSDS coordinator (see Section 8.10.5).

Ensure that contractors and subcontractors performing potentially hazardous work in SNF Project facilities are obligated to comply with the mandatory occupational safety and health standards of DOE Order 5480.4.

8.3.10 U.S. Department of Energy, Richland Operations Office  According to DOE Order 5480.1B *Environmental Safety and Health Program for Department of Energy Operations*, the heads of DOE field offices are responsible for ensuring that all operations under their jurisdiction are carried out in a manner consistent with sound safety and health practices and in accordance with all applicable DOE regulatory requirements. This responsibility includes the requirement by DOE to appraise the hazardous material protection program and the entire facility in accordance with DOE Order 5482.1B *Environment Safety and Health Appraisal Program* and other applicable regulatory requirements.

8.3.2 Staffing Levels and Qualifications

The SNF Project occupational safety and health organization is staffed by qualified and experienced personnel who are trained in the anticipation, recognition, evaluation, and control of occupational safety and health hazards in the work environment in accordance with Section 1912 of the SNF Project S/RID (HNF-SD-SNF-RD-001). Formal qualification criteria are established to define the experience, education, capability, and training required to perform a designated position. There are established requirements for periodic performance reviews, evaluations, and updates of qualification and requalification criteria. As indicated in Chapter 170, minimum education and experience requirements for management and workers meet the requirements of Section 4.0 of the SNF Project S/RID. SNF Project management determines the basic educational and experience criteria for the occupational safety and health staff positions (also see Section 8.5).

The occupational safety and health organization determines the basic educational and experience criteria for employees assigned to industrial hygiene staff positions. Industrial hygiene personnel must have a Bachelor of Science degree and industrial hygiene safety experience. Area industrial hygiene technicians must have a high school diploma. Certification (certified industrial hygienist or occupational health and safety technician by the American Board of Industrial Hygiene or certified safety professional by the Board of Certified Safety Professionals) is highly desired. Requirements for being a fully qualified safety professional include completion of a contractor qualification program and a minimum of two years related experience.
8.4 ALARA POLICY AND PROGRAMS

While there is no established formal SNF Project as-low-as-reasonably-achievable (ALARA) program for nonradiological hazardous materials, the SNF Project has expanded the classic concept of ALARA principles (i.e., minimization of radiological exposures) to the application of exposure minimization for hazardous substances and conditions. SNF Project policy is to limit human exposures to radiation and hazardous materials to levels below federal and state regulatory limits and further to levels ALARA. Therefore, ongoing programs promote reduction of exposures to chemical hazards, ionizing and nonionizing radiation, and other physical hazards through education and worker involvement. Many of the same principles and work practices that are specified in the ALARA program described in Chapter 7.0 also are used to minimize exposures to nonradiological hazardous materials.

The hazardous material exposure control program, discussed in Section 8.6, and the hazard communication program, discussed in Section 8.10, are provisions of the SNF Project occupational safety and health program directed at limiting exposures to hazardous materials to ALARA levels. Engineering and administrative controls, work practices, and personal protective equipment are instituted to reduce and maintain personnel exposures to or below the PELs for substances regulated by 29 CFR 1910 Subpart Z or TLVs from ACGIH whichever is more conservative.

The following work practices are used for hazardous materials protection at the SNF Project facilities:

- Prejob planning
- Job monitoring
- Post-job reviews
- Chemical exposure control
- Lessons-learned reviews
- Cost–benefit analysis worksheets
- Project and task planning, design, and reviews
- ALARA awareness
- Training

Chemical exposures in the workplace are further controlled by:

- Requirements of the purchasing process for chemical agents
- Requirements for handling of chemicals
- Chemical monitoring
- Hazard communication program
- Substituting nonhazardous or less hazardous chemicals
- Records and reporting

A summary for most of the above contributions that each practice provides to implementation of ALARA concepts is provided in Table 8-1.
Table 8-1  As Low As Reasonably Achievable Implementing Practices and Job-Related Activities (2 sheets)

<table>
<thead>
<tr>
<th>Support activity</th>
<th>Description/main features of activity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Prejob planning</td>
<td>Job hazard analysis or ISA forms are used to identify potential hazards</td>
</tr>
<tr>
<td></td>
<td>Qualitative hazard assessments are performed to follow up potential hazards</td>
</tr>
<tr>
<td>Job monitoring</td>
<td>Used when historical data is insufficient to predict exposure</td>
</tr>
<tr>
<td></td>
<td>Ensures that exposures are quantified</td>
</tr>
<tr>
<td></td>
<td>* Ensures that job follows plans and procedures</td>
</tr>
<tr>
<td></td>
<td>* Ensures awareness of known or potential hazards addressed through engineered and administrative controls</td>
</tr>
<tr>
<td></td>
<td>* Includes recording of job events that can be used to evaluate job during post job review</td>
</tr>
<tr>
<td>Post job review</td>
<td>Validates quantitative data provides evidence of actual exposure levels</td>
</tr>
<tr>
<td></td>
<td>Identifies problems encountered during the job, including evaluation of hazardous materials controls</td>
</tr>
<tr>
<td></td>
<td>Provides input to lessons learned for future jobs and tasks</td>
</tr>
<tr>
<td></td>
<td>* Ensures use and adequacy of PPE for planned tasks</td>
</tr>
<tr>
<td>Chemical exposure control</td>
<td>Requirements established and implemented for the purchasing process for chemical agents</td>
</tr>
<tr>
<td></td>
<td>Requirements established and implemented for handling of chemicals</td>
</tr>
<tr>
<td></td>
<td>Implementation of chemical monitoring</td>
</tr>
<tr>
<td></td>
<td>Establishment and implementation of a hazard communication program</td>
</tr>
<tr>
<td></td>
<td>Substitution of nonhazardous or less hazardous chemicals</td>
</tr>
<tr>
<td></td>
<td>Establishment of an effective records and reporting program</td>
</tr>
<tr>
<td>Lessons learned review</td>
<td>Ensures awareness and incorporation of information from previous similar work and activities</td>
</tr>
<tr>
<td>Project/task planning design and</td>
<td>Periodic review of current procedures minimizes hazardous material exposure by using known methods for</td>
</tr>
<tr>
<td>review</td>
<td>* Containment and controls</td>
</tr>
<tr>
<td></td>
<td>* Use of appropriate experienced and knowledgeable safety staff ensures that hazardous materials design</td>
</tr>
<tr>
<td></td>
<td>* Design criteria contain the ALARA concepts</td>
</tr>
<tr>
<td>ALARA awareness and hazard</td>
<td>Hazard communication training reinforces ALARA concepts through communications to workers</td>
</tr>
<tr>
<td>communication program</td>
<td>Uses communication tools (e.g., MSDSs, product labeling, chemical inventories training)</td>
</tr>
</tbody>
</table>

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The occupational safety and health program includes the following additional provisions:

- Exposures to hazardous materials are controlled within the PELs established by OSHA and the TLVs established by the ACGIH. Where a lower published exposure limit exists, the SNF Project will use that limit, according to the hierarchy published in DOE Order 5480.10.

- Routine work hazards are identified and communicated to workers through morning meetings, safety meetings, and routine worker–supervisor communication.

- Workers are involved with both the prejob planning process and prejob walkdowns whenever possible.

- Prejob briefings are held for nonroutine work to communicate hazards and controls to workers who will perform the task.

- Hazard information is communicated to the affected workers for routine work using an existing ("standing") JHA, JSA, automated JHA, approved procedure safety meeting, or routine worker–supervisor work discussion.
Nonroutine tasks require generation of a job-specific JHA, JSA or a computer-based qualitative JHA or the use of a permit, approved procedure, or prejob safety meeting to address the anticipated hazards and control measures of the task.

A hazard communication program provides workers and management with up-to-date accurate and timely information on hazards in their workplace and ways to reduce those hazards.

The occupational safety and health program is integrated into the SNF Project assessment process, which includes independent assessments, audits, and surveillances, management assessments, management workplace walkthroughs and on-the-job supervision.

Workplace hazard monitoring is conducted to detect physical or biological hazards or for any operation that can reasonably be expected to result in worker exposures equal to or greater than the action level for that chemical or substance.

An automated job task analysis is performed to support collection of the data necessary for a risk-based approach to medical qualification and monitoring. Information collected represents a compilation of hazards and exposures associated with routine work activities as well as hazards associated with nonroutine work activities that can be predicted or anticipated. Hazard and exposure information for nonroutine activities that cannot be predicted or anticipated are identified through JHAs, JSAs, and automated JHAs and incorporated into approved procedures.

Qualitative exposure assessments are conducted to assess the potential for occupational exposure of workers to chemical, physical, and biological hazards to evaluate and recommend control measures for potential hazards to communicate information regarding chemical, physical, and biological exposure hazards and controls to line management, and to document the qualitative exposure assessment results.

Chemicals are properly handled and stored.

Nonhazardous or less hazardous chemicals are substituted whenever possible.

A program will be implemented to minimize occupational exposure to chemical carcinogens as required by Section 1912 of the SNF Project S/RID (HNF-SD-SNF-RD-001).

Safety plans or standard operating procedures will describe the use of chemical carcinogens and the procedures used to contain exposure. The occupational safety and health organization will review and approve these documents.
SNF Project implementing procedures will establish emergency actions involving chemical carcinogens (e.g., cleanup of spills or accidental releases).

Notices of hazardous chemicals and physical and biological agents known to be present in the work area are posted.

Chemical agents that are brought into a facility are controlled by requiring that the hazards associated with the agents be determined before purchase, by requiring MSDSs from all manufacturers or suppliers, and by monitoring the workplace.

Any hazardous material or hazardous process that creates a potential worker exposure or oxygen-deficient atmosphere is reviewed and evaluated. This includes evacuating and reinerting the multi-canister overpacks.

Workers have access to and are trained to use MSDSs.

The SNF Project oversees contractor and subcontractor activities at SNF Project facilities to minimize exposures to hazardous materials through worksite inspections following the requirements of the SNF Project S/RID. Subcontractors are required to identify any chemicals or products that they will furnish to complete their work and are required to provide MSDSs for those chemicals prior to use.

DOE Order 5480.10, Revision 0, Section 9b, 'Introduction,' describes an industrial hygiene program designed to preserve employee health and well-being. Ergonomics is proposed as either an environmental factor or stressor under this program (e.g., body position in relation to task, repetitive motion, and mental or physical fatigue). For purposes of the SNF Project, the occupational safety and health concept of ergonomics is captured in Chapter 13 of Human factors engineering includes ergonomics as design criteria to consider in the project design. Human engineering includes human capabilities and limitations that must be considered in the design, with trade-offs appropriately considered as applicable based on such factors as safety risks, costs, benefits, and expediency.

The applicable ergonomics requirements of DOE Order 5480.10 will be included and developed further in the human engineering program plan (SNF-4399).

INEL-95/0117, Human Factors Engineering Checklists for Application in the SAR Process, contains a series of 19 human factors checklists that support analyses of DOE reactor and nonreactor facilities and activities. Many of these checklists contain ergonomics criteria (e.g., physical accessibility) and are included in the SNF Project human factors engineering analyses and findings reports.
8.5 HAZARDOUS MATERIAL TRAINING

Access to SNF Project facilities is limited during fuel handling operations. During staging and interim storage modes, the SNF Project facilities are expected to require only periodic monitoring by SNF Project workers. Therefore, only a few selected SNF Project workers assigned such monitoring duties are required to have specialized hazardous material training. Workers will perform only those tasks for which they have received the proper training. As the SNF Project mission changes, SNF Project management will review the training requirements and modify worker training requirements accordingly.

The SNF Project has an established hazardous material training program that complies with the requirements of 29 CFR 1910 and applicable provisions of DOE Order 5480.10. Occupational safety and health staff are trained in anticipating, recognizing, evaluating, and controlling hazards in the work environment. All workers are provided with initial general training on recognizing and minimizing job-related hazards. The process by which training is developed is discussed in Chapter 12.

SNF Project management provides training, professional education, and certification opportunities necessary to support, maintain, and enhance industrial hygiene staff proficiency to meet or exceed DOE industrial hygiene training objectives and goals. Program elements include needs assessment, definition of training objectives, training plans and implementation and evaluation of program effectiveness. Training is appropriate for the assigned tasks and level of responsibilities. Training also addresses hazards described in DOE-prescribed OSHA standards as well as those for which no PEL or other requirement has yet been established. The contractor integrates DOE-mandated training objectives into training programs as appropriate for each level of qualification for professional industrial hygiene staff, line management and supervision, and other workers.

The safety and health requirements of Section 19 of the SNF Project S/RID (HNF-SD-SNF-RD-001) ensure that all contracted or subcontracted activities are conducted in a safe manner. Significant activities involving safety are identified so that proper procurement or contract clauses will be included. Subcontractors will address the activities in their safety plans and safety professionals can assist in controlling risks. The SNF Project construction manager is responsible to ensure that subcontractor work groups are provided with the training, resources, and technical support necessary to perform their assigned safety and health duties. SNF Project implementing procedures provide information on implementation of the training requirements.

SNF Project facility managers are responsible for ensuring that workers assigned to any task involving hazardous materials are trained in the safety and health hazards associated with such hazardous materials. Workers receive on-the-job, area-specific training for the chemical hazards they work with that are present in their work areas. This training includes the following topics:

- How to recognize the presence or release of hazardous chemicals in the work area
Safe work practices for the chemical agents present in the workplace and work area

Controls that have been implemented to ensure exposures are reduced to below OSHA-established limits or limits set by the ACGIH

Information about the physical and health hazards of any chemicals present in the work area

Procedures and safety requirements that workers can use to protect themselves from hazards

Instructions for use of personal protective equipment in accordance with 29 CFR 1910.132 "General Requirements"

Response to chemical emergencies, such as a release, fire, or explosion

Specific training courses that cover the above topics include

- General Employee Training. This training provides the worker dealing with hazardous material and/or waste with the fundamentals for use and disposal of hazardous material. Information is provided on federal and hazardous communication programs, the use of MSDSs, labeling, requirements for manufacturers, Hanford Site labels, general chemical handling, safety, health effects, waste minimization, waste management regulations, the SNF Project's waste management system, and an overview of mixed waste requirements. Hazard communication and waste management awareness training introduces workers to federal laws governing chemical safety in the workplace. This safety training is mandated for general industry by 29 CFR 1910.1200, "Hazard Communication."

- Advanced General Employee Training. This training series provides the advanced general worker dealing with hazardous material and/or waste with detailed information for use and disposal of hazardous material. Course material covers various topics including regulatory requirements, waste characterization, waste storage disposal, packaging of dangerous or mixed waste, and preparation of packages for shipping.

- Hazardous Waste Site Supervisor/Manager Training. This training provides safety management training required for supervisors and managers of hazardous waste sites or operations covered by 29 CFR 1910.120, "Hazardous Waste Operations and Emergency Response."

- Hazardous Waste Operations. This training is a 24- or 40-hour hazardous waste operations course (depending on specific job requirements) that meets the requirements of 29 CFR 1910.120.
SNF Project facility managers are responsible for determining the mandatory task-specific training requirements for their workers. Examples of training include:

- **Confined Space Entry** Confined-space entry training is mandatory for permit-required, confined-space entry supervisors, attendants, and entrants. The course covers basic awareness of confined space hazards, entry permits, recognition of air contaminant over-exposure or oxygen deficiency symptoms, and emergency response requirements.

- **Fall Protection** Fall protection training is designed for workers protected by fall prevention systems and for users of personal fall arrest systems. The course covers a program overview, fall protection requirements at the SNF Project facilities, hands-on equipment inspection, and proper use of a variety of personal fall arrest systems.

- **Behavior-Based Safety Training** Behavior-based safety training contains elements of the VPP, conduct of operations, ALARA program, and total quality.

- **Asbestos Control** Asbestos control training covers recognition and identification of asbestos-containing materials and a review of applicable regulations, asbestos exposure hazards, chronic health effects, protective clothing, respiratory protection, waste disposal, and abatement techniques.

- **Hazards Recognition** Hazards recognition training provides OSHA-based training for recognizing hazards for general workers and supports requirements of the OSHA VPP.

- **Manager Safety Training** Manager safety training provides training in basic emergency medical care with emphasis on Hanford Site-specific implementation of safety standards and the VPP and includes discussion of relevant health and safety issues.

- **Medic First Aid** Medic first aid training provides instruction in basic emergency medical care with emphasis on developing ten basic skills: primary assessment, one-rescuer adult cardiopulmonary resuscitation, obstructed airway management, bleeding control, shock management, illness assessment, injury assessment, set-up and barriers, rescue breathing, and circle of care.

- **Noise Control** Noise control training provides OSHA training that meets requirements for workers exposed to excessive noise in the workplace.

The SNF Project training program establishes mandatory retraining for certain safety-related topics and procedures. It is the responsibility of SNF Project facility management to ensure workers are retrained within the time allowed by the course requirements for any specific training subject. The retraining sequence for each course is published in the SNF Project training program and is tracked on the computer-based training matrix database. As stated in...
Section 8.4 many of the same principles and work practices that are specified in the radiological ALARA program are used to minimize exposures to nonradiological hazardous materials. ALARA training will be provided to workers involved in minimizing hazardous materials. In addition to the above training, workers at SNF Project facilities will take training courses required for working at a Comprehensive Environmental Response Compensation and Liability Act (CERCLA) of 1980 site in accordance with 29 CFR 1910.120. Subcontractors are required to have training equivalent to that required for SNF Project staff members.

8.6 HAZARDOUS MATERIAL EXPOSURE CONTROL

Occupational exposures to hazardous materials and the spread of hazardous material contamination are controlled by a combination of engineered, operational, and administrative controls and by the use of personal protective clothing and equipment. Every effort is made to eliminate the hazards first through positive means or engineering controls before using administrative controls and personal protective equipment. Administrative controls include establishment of work zones to minimize the spread of contamination and implementation of procedures, training, and technical safety requirements. Engineering controls include confinement ventilation equipment layout, and high-efficiency particulate air filter placement.

HNF-MP-003 describes the philosophy and policies that provide reasonable assurance that the SNF Project facilities and activities are designed, constructed, and operated in a manner that ensure protection of workers, the public, and the environment. The key safety management concepts regarding hazardous materials are based on the following safety concepts:

- **Defense-in-depth** — Design facilities to provide multiple levels of defense against undue exposures to the workers and the public.
- **Maintain compliance with the authorization envelope** including using the least toxic chemicals or products that will provide the desired result.
- **Minimize exposures** — Keep exposures ALARA.
- **Hazard and/or safety analyses** — Define the hazardous aspects of the activity and the features needed to render the probability of inadvertent exposure of workers and the public extremely low.
- **Clear delineation of safety responsibility** — Define and exercise responsibility, when using hazardous materials, to ensure the safety of workers and the public and to protect the environment.

As described in HNF-MP-003, the following safety practices and requirements embody these fundamental safety concepts.
Prevention requirements that pertain to hazard analysis prevent undue exposures whether from normal or abnormal conditions attendant to the work activity or from unusual but credible events. Hazard analysis is also used to identify chemical incompatibilities and reactions that must be controlled.

- Use of materials that have the least adverse impact on workers, the public, and the environment but will provide the desired results.

- Preservation requirements to preserve the designed-in capability of the structures, systems, and components that are important to nuclear safety and to protection of the environment.

- Mitigation requirements that reflect possibilities for operational mishaps, human- or nature-caused and the emergency response capabilities needed to regain control and mitigate the consequences of dispersed hazardous material if released beyond the immediate design barriers.

- Management requirements that address the need for detailed procedures and trained and qualified workers to integrate, manage, and execute the safety functions.

An independent review is conducted to ensure that the requirements selected as the basis for new SNF Project facilities provide an acceptable level of worker safety, public health, and safety and protection of the environment.

### 8.6.1 Hazardous Material Identification Program

The industrial hygiene staff will identify and document existing and potential occupational health hazards in accordance with requirements of DOE Order 5480 10. This is performed through:

- Knowledge and assessment of the operations
- Periodic walkthrough surveys
- Information provided by inter-organizational communications
- Review of proposed projects, facilities, engineering plans, and specifications
- Maintenance of a hazards inventory or tracking system

Hazardous materials in the workplace are identified by several mechanisms including:

- Hazard communication program
- Hazardous materials and hazardous waste labeling
- Inventory of chemicals
- Facility modification assessments
- General employee health and safety training
- Hazardous materials characterization activities
- Ongoing and continuous source monitoring
- Worker and work area monitoring
- Potential hazard and accident analysis

Prejob planning, including the performance of a JHA, JSA, or automated JHA ensures that potential health and safety hazards are identified and controlled. The JHA, JSA, or automated JHA is performed through the combined efforts of a team (management and workers) finding potential hazards and then eliminating or minimizing them before the job starts. This process results in determining the best way to perform a job safely for both routine and nonroutine work activities. Involvement of workers and technical support organizations (e.g., occupational safety and health and radiological) in both hazard identification and hazard control components is essential to the hazard analysis process. All work activities will receive adequate planning so that potential hazards and accidents are identified and specific precautions are applied. The following steps are included in the JHA, JSA, or automated JHA process:

- Determine the scope and general description of activities to be performed
- Break the job or task down into separate steps or components in the order of planned occurrence
- Identify and assess all known or potential hazards associated with the job or task and areas where workers may be exposed to hazardous conditions
- Determine effective measures to remove or control the potential hazard
- Document and secure approval of appropriate organization functions including Occupational Safety and Health, Fire Protection, Nuclear Safety, Radiological Control, and Environmental Protection
- Participate in a prejob safety meeting using the JHA and/or JSA and other safety-related material relevant to the job or task
- Perform the job or task in accordance with the approved JHA and/or JSA

As needed, the JSA is supplemented by a qualitative exposure assessment of the operations activities, and tasks that present a significant potential for occupational exposure to chemical, physical, and biological hazards. The qualitative exposure assessment is intended to assess the potential for occupational exposure of workers to chemical, physical, and biological hazards, evaluate and recommend control measures for potential hazards, communicate information regarding chemical, physical, and biological exposure hazards, and controls to workers and line management, and document the qualitative exposure assessment. A qualitative exposure assessment includes provisions for conducting occupational safety and health walkthroughs of operations and activities to identify exposure agents and their sources, identify where and when in the operation or activity (e.g., step, stage, location) exposures could occur, and identify control measures.
As discussed in Chapter 30, an assessment team conducted a hazard analysis to identify potential hazards to the SNF Project facility worker, collocated worker, the public, and the environment. Use of the hazard analysis study to identify worker safety concerns and potential accidents satisfies the process safety management guidelines in 29 CFR 1910.119 "Process Safety Management of Highly Hazardous Chemicals," relative to evaluating hazards of a process.

Process safety management is an integrated approach to hazard identification and risk management.

The SNF Project also has a well-defined, systematic approach to performing industrial safety and health comprehensive baseline hazard assessments. Comprehensive baseline hazard assessments are conducted in accordance with SNF Project implementing procedures to identify workplace hazards, evaluate controls and their effectiveness, and to verify effective implementation of applicable safety requirements. Comprehensive baseline hazard assessments and abatement of identified hazards are performed with direct worker involvement. Results of assessments are submitted to the responsible operations managers and their assigned workers.

Chapter 30 describes the evaluation methodology for identifying hazards. To identify potential SNF Project hazards, the processes and activities that will take place within each area of SNF Project facilities were determined. A standardized hazardous material and energy source checklist was used to group potentially hazardous materials and energy sources as they were identified in each area. The hazards were further identified by determining hazardous conditions resulting from the hazardous material or energy source, establishing accident initiators or causes, and defining resultant accident sequences.

Worker safety features are an integral part of facility design. The major features of worker protection are identified and categorized by hazard. The hazard location, potential accident, and consequences are identified along with protective features, including passive, active, and administrative features. The SNF Project uses defense-in-depth engineered and administrative features that provide layers of protection to workers from those hazards. Defense-in-depth engineered features include those that function as:

- Barriers to contain uncontrolled hazardous material or energy release
- Preventive systems to protect those barriers
- Systems to mitigate uncontrolled hazardous material or energy release upon barrier failure

8.6.2 Administrative Limits

SNF Project policy requires that personnel exposure to hazardous materials be maintained at less than the OSHA and ACGIH airborne concentration limits. Personnel exposures to hazardous materials are controlled within the PELs and TLVs. Where OSHA and ACGIH have not established limits, recommendations by other agencies such as the National Institute for...
Occupational Safety and Health and the American Industrial Hygiene Associations, are considered in developing an exposure limit. Guidelines on evaluation and control are developed using the best available information and the professional judgement of the safety and health specialist. In the absence of any recommended limits, nonroutine exposure limits are prescribed on a case-by-case basis for the particular hazard. The occupational safety and health staff will establish these limits conservatively using a combination of factors derived from professional experience and knowledge, available data, and an assessment of the particular situation.

8.6.3 Occupational Medical Programs

The Hanford Site occupational medical contractor is responsible for providing an effective, appropriate, medical surveillance program, including maintaining all associated records and documents. Bioassay requirements are specified by the occupational safety and health organization for workers who have the potential for exposure to heavy metals, carcinogens, or compounds for which the ACGIH has specified biological exposure indices. Bioassay and medical monitoring programs are responsive to the requirements of 29 CFR 1910.120 and 29 CFR 1910 Subpart Z.

The automated EJTA provides the primary mechanism to ensure that workers have the appropriate medical qualification, training, and exposure monitoring based on their assigned job functions and the hazards to which they may be exposed. The EJTA also:

- Satisfies specific Americans with Disabilities Act of 1990 and Fitness for Duty data needs
- Identifies the need for additional employee exposure assessments and monitoring data
- Aids in determining the necessary health and safety training

The job task analysis in conjunction with the automated JHA and exposure monitoring and reporting provides the primary data input components to the Hanford Site occupational health process. In addition to providing essential data for medical qualification and monitoring, the Hanford Site occupational health process effectively supports other occupational medical qualification monitoring, and evaluations and preplacement, voluntary periodic, return to work and termination health examinations, which are specified by the SNF Project S/RID (HNF-SD-SNF-RD-001).

The specifics of the occupational medical program provided for the workforce are described in the memorandum of agreement between the occupational medical contractor, and Fluor Daniel Hanford, Incorporated. The occupational medical contractor schedules workers for medical qualification examinations and medical monitoring based on data provided through the EJTA.
Removes workers from medical program placement when warranted based on EJTA information and other relevant medical information

- Reports results of medical examinations and monitoring to workers and line management
- Maintains medical records in accordance with the applicable OSHA and DOE requirements
- Provides first aid services, including diagnosis and treatment of non-life threatening events
- Performs treatment following accidents, injuries or illnesses
- Conducts worker medical examinations and evaluations in accordance with the S/RID
- Provides fitness for duty on workers for all conditions that may influence performance or work suitability
- Provides medical input into the Fluor Daniel Hanford Incorporated Environmental, Safety Health and Quality organization procedures
- Conducts worksite assessments

All SNF Project workers have the right to review their EJTA, JHA they are working or have worked on and their own medical records maintained by the occupational health contractor

**8.6.4 Respiratory Protection**

The SNF Project is committed to minimizing the inhalation of air contaminated with dusts, mists, fumes, gases, vapors, toxic substances and radionuclides in accordance with the requirements of Sections 19 3 2 and 19 4 1 of the SNF Project S/RID (HNF-SD-SNF-RD-001) which provide hazards controls and health examinations. These cover various hazardous materials, personal protective equipment, environmental controls, materials handling and storage, medical and first aid, work place inspections for hazardous conditions and other similar controls. The primary means of achieving this goal is to prevent or mitigate the hazardous condition at the source. Every reasonable effort is made to achieve this objective by using engineering and administrative controls such as enclosures, ventilation or process modification in accordance with ANSI Z88.2-1992 *Respiratory Protection*. Personal protective equipment requirements in accordance with 29 CFR 1910 are stated in job planning documents including JHAs, JSAs, maintenance procedures, operating procedures, and job-specific permits. These job planning documents include the consideration of normal, abnormal, and accident conditions.
Assessments are performed of work areas and work activities by the occupational medical contractor in accordance with Section 19 4 1 in the SNF Project S/RID (HNF-SD-SNF-RD-001) to determine the presence or likelihood of hazards. This will include a review of materials, processes, and procedures with an emphasis on physical, chemical, and biological hazards. Frequency of worksite visits is determined by the occupational medical director taking into account factors such as the size of the workforce, number and types of operations, nature and amounts of physical, chemical, or biological agents, accident and incident rate, and occupational illness and disability rate.

Prior to the health examinations, the SNF Project management will provide a summary of potential exposures to hazardous agents or tasks and any worksite exposures in excess of OSHA and DOE PELS pertaining to the employees to be examined. Recommendations will be provided to management for corrective action or preventive measures.

The occupational health examiner will perform a number of medical evaluations in accordance with DOE Order 5480 8A, including:

- Preplacement evaluations
- Medical surveillance examinations and health monitoring
- Qualification examinations
- Return-to-work health evaluations
- Termination health evaluations
- Medical monitoring

Section 19 3 2 (29 CFR 1910 134) and Section 19 4 1 in the SNF Project S/RID (HNF-SD-SNF-RD-001) provide the requirements for the respiratory protection program for radiological and nonradiological hazards, which will be implemented by SNF Project implementing procedures. Also see Section 8 6 1, which presents the details for identifying potential hazardous materials in the workplace. For nonradiological hazards, industrial hygienists will:

- Evaluate hazards and select respiratory protection to be used by workers for SNF Project activities involving hazardous materials
- Evaluate hazards and conduct hazardous material exposure assessment as required to ensure adequacy of respiratory protection
- Support line management, respirator issuers, and respirator wearers in ensuring the proper application control, issue, use, cleaning, maintenance, repair, and care for respirators/cartridges at the worksite
- Maintain applicable documentation and records
- Serve as initial point of contact for any issue resolution
- Order cartridges/canisters as needed for specific applications involving hazardous materials

### 8.6.5 Bioassay Program

SNF Project implementing procedures describe the process for implementing the internal dosimetry program and bioassay program, including specific actions and responsibilities of Fluor Daniel Radiological Protection, major subcontractor organizations and Battelle Pacific Northwest Dosimetry Operations in accordance with Title 10, *Code of Federal Regulations*, Part 835, 'Occupational Radiation Protection,' Section 835.402 'Individual Monitoring' (10 CFR 835), and HSRCM-1, *Hanford Site Radiological Control Manual* The need for a baseline or routine bioassay examination may be triggered by a potential for internal exposure in which prejob planning or the radiation work permit indicate that a bioassay examination is required for:

- New employee
- Rehire
- Vendor
- Visitor
- Subcontractor employee
- Employee returning from a leave of absence

In addition, special examinations are initiated if a worker is exposed to the following:

- Radioactive contamination on the worker or potentially in a wound
- Airborne contamination where the worker could have received greater than or equal to 10 derived air concentration hours of internal exposure (after calculating for respiratory protection factors)

If the indicators indicate contamination by transuranic radioisotopes (plutonium, americium) or if an uptake of transuranic isotopes is considered possible, an on-call occupational health contractor physician is notified that the test criteria and test results support use of diethylene triamine penta acetic acid therapy. Results of the testing in the form of internal dose evaluation reports (accident) routine results report (less than decisional levels), and routine results report (greater than investigative levels) is discussed with the manager and worker including the health and safety implications of the results. An occurrence report including test results, is developed and presented to the worker and DOE.

Workers may be restricted from radiological work for any of the following reasons:

- Failing to submit required routine bioassay samples
- Missing a scheduled in vivo test

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Experiencing a suspect or uncertain dose requiring further evaluation to assess
Any other assessed situation that could result in undetected internal dose

A report is prepared periodically (about annually) by Pacific Northwest National Laboratory that supplies information to support decisions made in assigning routine bioassay requirements in radiation work permit preparation. Reports consist of the following information:

- Identifies what routine bioassays are required for standard work activities
- Identifies exceptions to the bioassay criteria for standard work activities
- Contains characterization data or other determinations necessary to support the conclusions
- Applicable radiological work permit guidance
- May contain process flow charts, waste characterization or other available documents
- Applicable internal dosimetry technical basis guidance
- Considers what radionuclides are most likely to be dispersed in an accident (i.e., contamination that is resuspendable is more important for routine bioassay planning than material that is fixed and unlikely to contribute to a worker intake)
- Includes a discussion on the use of air sample analysis and tech smears of material likely to be resuspended

8.7 HAZARDOUS MATERIAL MONITORING

This section provides summaries of the hazardous material sampling and monitoring programs that are conducted internally and externally for SNF Project facilities in accordance with Section 19.3 of the SNF Project S/RID (HNF-SD-SNF-RD-001). Additional details of the program are presented in the following subsections.

8.7.1 Workplace Monitoring and Sampling Program

The SNF Project administrative procedures describe the process for implementing the workplace monitoring and sampling program. The objectives of the workplace monitoring and sampling program are to provide a rational basis for selecting appropriate levels of personal protective equipment and work practice controls, documenting personnel exposure levels and verifying that the implemented hazard control procedures are adequate and appropriate for actual facility conditions. SNF Project occupational safety and health manuals and procedures address
monitoring for workplace hazards. See Section 8.4 for a discussion of the SNF Project policies and programs for maintaining nonradiological hazardous materials. ALARA. Also see Section 8.6.1, which provides details on prejob planning, developing a JHA, JSA, or automated JHA, worker and work area monitoring, identifying potential hazards, involvement of workers and technical support organizations in hazard identification, and hazard control processes.

8.7.2 External Monitoring and Sampling Program

The SNF Project administrative procedures describe the process for implementing the external monitoring and sampling program. The principal function of the external monitoring and sampling program is to detect, quantify, evaluate, and, where possible, predict impacts associated with routine as well as accidental or unintended releases of radioactive as well as nonradioactive constituents to the environment. Monitoring programs also are directed at maintaining compliance with environmental permits from the Washington State Department of Health and with other applicable regulations. The monitoring programs are directed at atmospheric, physical and biological parameters both on and off the site. As required by Section 19.3.1 of the SNF Project S/RID (HNF-SD-SNF-RD-001), the industrial hygiene staff is responsible to determine the type and frequency of periodic monitoring that is performed to ensure the maintenance of satisfactory conditions. The industrial hygiene staff also determines the type and frequency of periodic monitoring, including continuing adequacy of controls, need for additional controls or recommendations for maintenance or reemphasis of administrative controls.

Section 20.3 of the SNF Project S/RID (HNF-SD-SNF-RD-001) presents requirements for environmental monitoring, surveillance, and inspection activities applicable to the SNF Project. Environmental surveillance includes sampling and analyzing environmental media to detect and quantify potential contaminants and to assess their human health significance. This includes sampling of air, surface water, soil, and vegetation. The environmental monitoring activities are performed independently of the occupational safety and health organization's monitoring activities by the SNF Project environmental monitoring organization.

A preoperational environmental monitoring study will be initiated prior to startup of SNF Project facilities in accordance with Section 20.3 of the SNF Project S/RID (HNF-SD-SNF-RD-001). This will be completed between one and two years prior to startup so as to include any seasonal changes. This study will include the following:

- Characterization of existing physical, chemical, and biological conditions that could be affected
- Establishment of background levels of radioactive and chemical components
Characterization of pertinent environmental and ecological parameters

Identification of potential pathways for human exposure or environmental impact as a basis for determining the nature and extent of the subsequent routine operational and effluent monitoring and environmental surveillance programs.

An environmental monitoring plan will be developed for each SNF Project. This will contain rational and design material for the monitoring program, extent and frequency of monitoring and measurements, procedures for laboratory analyses, quality assurance requirements, program implementation procedures, and direction for preparation and disposition of the reports. The plan will be reviewed annually and updated as necessary.

Analytical models to perform dose evaluations will be appropriate for:

- Emission characterizations (e.g., gas, liquid or particle, depositing or nondepositing buoyant or nonbuoyant)
- Mode or release (e.g., stack or vent, crib or pond, continuous or intermittent)
- Environmental transport medium (e.g., air or water)
- Exposure pathway (e.g., inhalation, ingestion of food, water, or milk, direct radiation)

Information will be updated as necessary to document significant changes that could affect dose evaluations.

SNF Project implementing procedures will include the above and related requirements of Sections 20.4 and 20.5 of the SNF Project S/RID (HNF-SD-SNF-RD-001).

Section 20.6 of the SNF Project S/RID (HNF-SD-SNF-RD-001) present recordkeeping, reports, and notification requirements. Responsibilities and response actions to be initiated in the event of an accidental routine or nonroutine release of a solid, liquid, or airborne substance that includes radioactive hazardous or dangerous wastes, hazardous or extremely hazardous substances, polychlorinated biphenyls, or oil petroleum products are contained in SNF Project implementing procedures. Such releases are immediately reported regardless of the quantity except for small drips that can be immediately wiped up. The SNF Project is responsible for implementing actions to clean up and properly dispose of spilled material in accordance with the applicable federal, state, and local codes and regulations. This procedure also presents requirements for reporting chemical information pursuant to the Emergency Planning and Community Right-To-Know Act of 1986.
8.8 HAZARDOUS MATERIAL PROTECTION INSTRUMENTATION

Direct-reading instrumentation and other hazardous protection instruments are maintained in a pool. The instruments in the pool are treated as sitewide resources and are loaned to Hanford Site workers on an as-needed basis. The occupational safety and health organization assists the SNF Project facility industrial hygienists and technicians with preuse and postuse calibrations. A database is used to track the status and location of instruments being used, and status-indicating tags are used to control instruments requiring service and repair. Periodic, routine maintenance of the direct-reading instrumentation within the pool is the responsibility of the instrument pool organization. Calibrations are performed at an onsite laboratory or approved offsite laboratory using National Institute of Standards and Technology traceable standards. The types of standards (e.g., calibration gases, thermometers) and calibration technique, if performed onsite, meet manufacturer's specifications.

In addition to direct-reading field instruments, many types of personal monitors that require field sampling followed by laboratory analysis of samples are used. Field-sampling equipment is controlled in the same manner as the direct-reading instruments.

Typical direct-reading instruments available for workplace monitoring are listed in Table 8-2. SNF Project facility industrial hygienists and technicians will use the appropriate instrumentation for personnel protection. For several of the analytes listed, various types of instruments are available for sampling and monitoring.

The calibration and/or functional status of each direct-reading instrument is checked and documented by the person responsible for the equipment before and after each use. The calibration and/or functional status of each direct-reading instrument is checked and verified to be within established limits or the manufacturer's manual guidance. The results of this check are documented by the user daily and prior to use. All instrument field calibration data and monitoring results are recorded in a daily monitoring, if taken. As discussed in Chapter 14.0 each organization responsible for using measuring and testing equipment uses controlled procedures for calibration and control of that equipment. Any measuring and testing equipment used for acceptance testing, verification, or data collection are subject to these controls. These controls will provide for calibration and adjustment at specified intervals to maintain accuracy within necessary limits.

An SNF Project quality assurance program plan provides specific details that describe the activities and quality assurance controls for measuring and testing equipment for the SNF Project facilities. As discussed in Chapter 10.0, the SNF Project in-service surveillance program will contain provisions for testing and calibration, and control and calibration of measuring and testing equipment. Additional details of the program for the control and calibration of measuring and testing equipment are required by Section 10.1 of the SNF Project S/RID (HNF-SD-SNF-RD-001).
Table 8-2 Hazardous Chemical Monitoring Instrumentation for Personnel Protection

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<tr>
<th>Detector type</th>
<th>Analyte(s)</th>
<th>Sensitivity/ range</th>
<th>Specificity</th>
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**Toxics** = CO₂, HCN, CO, NO₂, H₂O, SO₂, and NH₃.

*The GC instrument is portable in principle but because of its weight it is typically used in a laboratory setting.

CL = chemiluminescence
ES = electrochemical sensor
FID = flame ionization detector
FL = fluorescence
GC = gas chromatograph
GF = gold film
IR = infrared
LEL = lower explosive limit (e.g., for oxygen, hydrogen, and methane)
PID = photo ionization detector
ppb = part per billion (range of 1 to 1,000 parts per billion by volume)
ppm = part per million (range of 1 to 1,000 parts per million by volume)
Safety and health specialists determine instrument use number of instruments and placement under normal and emergency conditions by drawing upon their skills and experience and by using data from a qualitative hazard analysis or baseline hazard assessment. The requirements for hazardous material protection instrumentation at the SNF Project facilities are based on the following:

- Type of work being performed
- Contaminant to be sampled or monitored
- Regulatory requirements (PELs or TLVs) for the contaminant
- Worker exposure levels or concentrations of hazardous substances
- Results of periodic work condition evaluations verifying that the selected levels of protection and control measures are appropriate

Direct-reading instruments are used for personal and area monitoring while work activities are being performed. Personal breathing zones are sampled using sorbent or filter media to confirm that exposure levels are within the regulatory exposure levels.

### 8.9 Hazardous Material Protection Record Keeping

The SNF Project has an established document control and records management program that ensures records are maintained in compliance with Sections 18 and 181 through 183 of the SNF Project S/RID (HNF-SD-SNF-RD-001). The activities and controls for SNF Project records are presented in a SNF Project quality assurance program plan.

Health and safety data and information are collected, maintained, and retrieved in a systematic manner that ensures data quality. The SNF Project hazard communication program requires that all monitoring data written reports of field evaluations, recommendations for corrective actions, and closures of actions be recorded and retained pursuant to Section 196 of the SNF Project S/RID (HNF-SD-SNF-RD-001).

Records shall be (1) adequate and proper that are created and maintained for all activities under the contract and (2) established in accordance with a records management program that provides approval of all records dispositioned by the National Archives and Records Administration. An approved records disposition schedule will be developed for every records service.

As required by Section 1961 of the SNF Project S/RID (HNF-SD-SNF-RD-001), all employees or former employees have access to their personal safety, health, and medical records consistent with the *Freedom of Information Act* and the *Privacy Act of 1974*.
If requested by the worker, occupational monitoring data are reported directly to the worker monitored. As required by Section 1931 of the SNF Project S/RID (HNF-SD-SNF-RD-001), the SNF Project management will inform workers of the existence, location, and availability of workplace-related medical and exposure records. These records are maintained by the Fluor Daniel Hanford, Incorporated Environmental Safety, Health and Quality organization and are provided to a worker or worker's representative, within 15 days of a request. Distribution of personal exposure monitoring data is governed by Section 1931 of the S/RID.

Data resulting from occupational monitoring will be easily retrievable. Monitoring data shall be tabulated along with information on the location and operation monitored, the identity and job classification of the employees associated with the operations, estimated time-weighted average or short-term exposure levels, and a reference to the sampling and analytical methods used in accordance with Section 1961 of the SNF Project S/RID (HNF-SD-SNF-RD-001). Data generated to provide information on worker and public hazardous material protection are reviewed for adequacy by qualified staff authorized for release and use by management, and distributed to those workers who request the information. Occupational monitoring data are reviewed before inclusion in the Hanford Site occupational safety and health exposure database. Reviews consist of data validation (to ensure that appropriate monitoring methods were employed), confirmation that instruments used were within calibration specifications, and data transcription checks (to ensure that the data are free of technical and clerical errors). The data are interpreted by comparing the validated monitoring data with recognized national standards to assess compliance and to determine whether engineering, administrative, or personal protective controls are needed to protect workers from assessed hazards.

The SNF Project is responsible for recording and reporting recordable occupational illnesses and injuries in accordance with the requirements of Section 1922 of the SNF Project S/RID (HNF-SD-SNF-RD-001). Records of employee exposure to toxic materials or harmful physical agents will be maintained in perpetuity. Records of violations of DOE-prescribed OSHA standards noted during inspections will be maintained. Retention periods will be established in accordance with the SNF Project S/RID requirements.

The SNF Project will post DOE forms 5480 2 and 5480 3 or DOE forms EV-632 and EV-632S. Posting will be sufficient to permit employees to observe the information on the way to or from their places of employment.

8.10 HAZARD COMMUNICATION PROGRAM

The hazard communication program applies to the purchase, receipt, transportation, use, and storage of hazardous chemicals and products. Specific elements of the program include area definitions and postings, chemical labeling, chemical product lists, and MSDSs. A description of the program and worker responsibilities is contained in SNF Project implementing procedures.
An inventory of potential occupational health chemical, physical, and biological hazards in SNF Project facilities will be maintained by location and/or job category of users and indicate when the hazards were present. The industrial hygiene staff evaluation of potential health hazards will be documented in written reports and deficiencies will be identified with recommended corrective actions. These reports and responses by SNF Project management will be retained in accordance with the records management program.

The purpose of the hazard communication program is to ensure that SNF Project workers are informed of these hazards and are provided with adequate information and training to know and understand the use of the necessary protective measures. The primary goal of the program is to provide adequate information to allow these materials or agents to be used, handled, stored, and disposed of safely. SNF Project personnel are required to promptly notify the Hanford Fire Department of any hazardous material spills or incidents requiring response by the Hanford Fire Department's hazardous materials response team. The SNF Project also maintains an ongoing dialogue with the Hanford Fire Department so the fire department is aware of unique or special hazards related to the SNF Project. This dialogue is required by SNF Project implementing procedures.

The SNF Project purchasing organization will participate as requested by the industrial hygiene staff to ensure that potentially hazardous material or equipment being procured are adequately identified, evaluated, and controlled. In addition, they will ensure compliance with the mandatory industrial hygiene standards in Section 1932 of the SNF Project S/RID (HNF-SD-SNF-RD-001) and make provisions to allow the industrial hygiene staff to monitor compliance.

8.10.1 Hazard Posting in Work Areas

Chemical and physical hazard information is posted in each specific work area location within the SNF Project facilities to advise workers of the hazards that are present in the workplace. The posting includes flammable, corrosive, reactive, toxic, and carcinogenic chemicals that are used or stored within the area. Informational postings also include other workplace hazards, such as physical or biological hazards. In addition to hazard warnings, hazard postings provide information on the location of MSDSs and spill cleanup materials. It is the responsibility of each worker to observe the hazard postings and to comply with the indicated requirements.

8.10.2 Chemical Management

A chemical management program has been established as required by the integrated environment, safety, health, and quality management system. Objectives that are achieved by this program include the following:

- Assure compliance with applicable regulatory and statutory requirements.
- Protect the worker, public, and environment
- Implement a consistent approach to the management of chemicals among the Project Hanford Management Contract Team and its subcontractors
- Incorporate the chemical management system requirements agreed to by the Hanford Site major prime contractors

An SNF Project implementing procedure has been developed that applies to all chemical management operations associated with the acquisition, storage, use, transportation, and final disposition of chemicals. This procedure also includes the Project Hanford Management Contract Team roles and responsibilities and key implementing procedures that establish requirements for chemical management. Chemical management requirements will be applied on a graded approach.

8.10.3 Chemical Labeling

The SNF Project hazard communication program will require that the manufacturer's labels and appropriate warnings be legible on all purchased hazardous chemical primary containers. The manufacturer's label on the original chemical container will not be removed or defaced. As a minimum, the manufacturer's label indicates the manufacturer's name and address, identity of the hazardous chemical, associated health and safety hazards, product or chemical name, appropriate hazard warnings, and special precautions. If a container is inadequately labeled, the label is corrected immediately by the person discovering the situation unless the contents are unknown. If the contents are unknown or improperly or inadequately identified, the contents are identified before use or disposal.

If a hazardous chemical is transferred from a primary container to another (secondary) container, the secondary container is labeled with a Hanford Site hazard label in accordance with established procedures. The Hanford Site label shows the MSDS number and numerical hazard rating. The Hanford Site label includes the name of the chemical and the product or trade name. Hazards are color coded as follows:

- Health hazard – blue
- Fire hazard – red
- Reactivity hazard – yellow
- Other hazards – white

The hazard severity identification includes:

- 0 – minimal or no hazard
- 1 – slight hazard
- 2 – moderate hazard
- 3 – serious hazard
- 4 – severe hazard
Chemical materials transferred from the original container into smaller, portable containers are not required to be labeled as long as their subsequent use is limited to the immediate use of the worker who performs the transfer.

8 10 4 Chemical Product List

All chemical products stored or used in SNF Project facilities are listed on the facility-specific chemical product list. The list identifies the product by providing the following information:

- Chemical name
- Site medical contractor-assigned MSDS identification number
- Manufacturer's number
- Determination of whether the product is a carcinogen
- Determination of whether the product, if disposed of, is a regulated waste
- Date received or placed in inventory
- Locations approved for use or storage

The facility's chemical product list is maintained by the engineering organization. A copy of the chemical product list is kept with the facility's MSDS files. Any chemicals not on the list are reviewed before they are ordered to determine whether a suitable substitute product could be bought that will perform the same function while reducing potential exposure to hazardous chemicals. To ensure that disposal requirements are known in advance of purchase of a new chemical, a waste summary and a waste specification record or hazardous waste disposal analysis record is prepared and approved. Procurement of a known or suspected carcinogen or other hazardous product will require justification, in writing, describing the reason a less hazardous chemical cannot be used. Training and information is provided for those using and handling a new chemical before the chemical is brought into the facility.

Chemical tracking is performed by the occupational safety and health organization to provide information about the storage location, product name, manufacturer, chemical constituents, physical state, container description, and total quantity for all chemical products stored in facilities that maintain an inventory of hazardous chemicals. Regular inspections are made of chemical storage areas and facilities to ensure container integrity, compatibility, adequate separation, temperature, moisture, and humidity control. Outdated material is disposed of properly. Any contaminated decomposed, or altered material is considered highly unstable and potentially hazardous. If such materials are found, (1) the container is not disturbed, (2) the area is isolated, and (3) the manager or building emergency director is notified. The manager or Building Emergency Director will contact the Industrial Hygiene organization and the Hanford Fire Department to arrange for proper handling and disposal in accordance with SNF Project implementing procedures.
8.10.5 Material Safety Data Sheets

An MSDS is required for each chemical and product on the hazardous chemical inventory listing. The MSDS provides a standard format for explaining chemical hazard information and is readily accessible to all workers. Each MSDS must provide the following information:

- Physical and chemical characteristics
- Physical hazards
- Health hazards signs and symptoms of exposure, and any medical conditions that are generally recognized to be aggravated by this chemical
- PEL and/or TLV
- Emergency and first aid procedures
- Applicable control measures such as engineering controls, work practices, or personal protective equipment
- Precautions for safe handling and use
- Name, address, and phone number of chemical manufacturer

The SNF Project procurement organization has the responsibility for ensuring that MSDSs are received with each shipment from the chemical manufacturer or supplier and submitted to the MSDS coordinator. Each chemical used in SNF Project facilities has an updated MSDS that is maintained by the MSDS coordinator. Outdated or superseded MSDS files are replaced by hard copies of the new MSDS. The MSDS is included in the work package for activities involving hazardous chemicals, and individuals performing the work are responsible for reviewing each MSDS in the work package before performing the work. The MSDS database available through the Hanford Local Area Network provides an inventory option so that facilities can track their chemical inventories through the MSDS request system. Using this system ensures that facilities are notified of any MSDS updates and/or new formulations.

8.10.6 Information and Training

As required by the OSHA hazard communication standard (29 CFR 1910.1200), workers and contractors are provided information concerning the hazardous substances to which they may be exposed and are made aware of exposure risks, hazards, personal protective equipment requirements, and emergency procedures associated with the hazardous chemical. This training covers an explanation of the SNF Project hazard communication program and how to find information regarding chemical hazards. All workers receive this training during Hanford General Employee Training.
During facility orientation, new workers are provided with facility-specific information regarding the specific hazards of chemicals that may be encountered during routine work. They are also informed of the location of chemicals in their work area, as well as the location and accessibility of the MSDS for any chemical in their work area.

Where work is nonroutine, a JHA is performed to systematically identify the hazards associated with the activities being performed and the areas in which a worker may be exposed to hazardous conditions. The hazard analyses identifies specific effective, safety measures that may be applied to eliminate or control the hazard. Prejob safety meetings are conducted to discuss the hazards with workers who may be exposed to the hazard during the course of their work. The prejob briefings address conditions in the workplace, hold points, and limiting conditions that may void the permits or work plans. The training also identifies engineering, administrative, and personal protective control mechanisms.

The SNF Project hazard communication program requires that facility managers inform Site contractor and subcontractor workers of chemical and physical hazards they may encounter. Contractors and subcontractors have the responsibility to inform facility managers of any chemical hazards and/or physical agents that they might bring onsite.

8.11 OCCUPATIONAL CHEMICAL EXPOSURES

No routine chemical processes are conducted within the SNF Project facilities. The multi-canister overpack process involves the use of an inert gas. Some chemicals, such as those used for equipment decontamination, may be used occasionally. However, the predicted annual exposures to workers from hazardous materials in SNF Project facilities are expected to be negligible for normal operation.

As discussed in Chapter 9.0, maintenance and monitoring activities may require the use of small amounts of activity-specific products that are not expected to contribute to occupational chemical exposures. However, in the event that additional hazardous materials are introduced into SNF Project facilities, occupational safety and health controls, such as engineered and administrative controls, as well as personal monitoring and sampling, are implemented to ensure that overexposure does not occur.

The Hanford Fire Department is the designated emergency response organization for all hazardous materials emergencies, including medical emergencies. The department's function is to mitigate and stabilize the emergency event. The department also is the designated rescue agency for the Hanford Site, including confined spaces, hazardous areas, cave-ins, trench rescue, building collapse, and high-angle rescues. In accordance with the SNF Project implementing procedures, all SNF Project employees are responsible for promptly notifying the Hanford Fire Department of any hazardous material spill or incident requiring response by the Hanford Fire Department. Hazardous materials response team members include any employee who becomes unexpectedly or extremely ill or injured.
8 12 REFERENCES


*Americans with Disabilities Act of 1990*, 42 U S C 12101, et seq


*Comprehensive Environmental Response Compensation and Liability Act (CERCLA) of 1980* 42 U S C 9601 et seq

DOE/EH-0433 1995 *U S Department of Energy Voluntary Protection Program*

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DOE Order 5480 4 *Environmental Protection Safety and Health Protection Standards* Change 4 (1993), U S Department of Energy, Washington, D C


DOE Order 5480 10 *Contractor Industrial Hygiene Program* U S Department of Energy, Washington D C


DOE Order 5482 1B *Environment Safety, and Health Appraisal Program* U S Department of Energy Washington, D C


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HNF-MP-003, Integrated Environment Safety and Health Management System Plan, Rev. 0, Fluor Daniel Hanford Incorporated, Richland, Washington

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HNF-PRO-1618, ALARA Management Commitment and Policy, Fluor Daniel Hanford Incorporated, Richland, Washington

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HNF-PRO-1620, ALARA Program Scope, Fluor Daniel Hanford Incorporated, Richland, Washington

HNF-PRO-1621, ALARA Decision-Making Methods, Fluor Daniel Hanford Incorporated, Richland, Washington

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HNF-PRO-1631, ALARA Training, Fluor Daniel Hanford Incorporated, Richland, Washington

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CHAPTER 9 0

RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT
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9.0 RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT

9.1 INTRODUCTION

This chapter describes the essential features of the radioactive and hazardous waste management programs that provide for safe control, collection, and handling of wastes generated during routine operations at Spent Nuclear Fuel (SNF) Project facilities. The overall radioactive and hazardous waste management programs and organization including objectives, plans, procedures, and training, are described. Radioactive and hazardous waste streams and sources are identified in the facility annexes to the SNF Project Final Safety Analysis Report (FSAR).

The SNF Project facility operations are not expected to produce appreciable amounts of hazardous wastes. The SNF Project facilities will produce miscellaneous industrial wastes, radioactive waste, and employee-generated trash. Miscellaneous industrial waste results from maintenance and repair of equipment, such as cranes and heating, ventilation, and air conditioning equipment. The major volumes of radioactive waste will come from K Basins operations and decommissioning. This chapter addresses the processes of packaging and transporting these waste categories.

The SNF Project waste management processes adequately protect the health and safety of the worker, the public, and the environment in a manner that meets as-low-as-reasonably-achievable criteria for radioactive and hazardous waste.

In those cases where policies, programs, and practices important to safe operation are described in detail in other documents, the information is summarized in this chapter and the documents are referenced. The detailed programs and procedures described in referenced documents may be changed without further US Department of Energy (DOE) approval to the extent that the changes do not constitute an unreviewed safety question as defined by DOE Order 5480.21 Unreviewed Safety Question.

9.2 REQUIREMENTS

The requirements that form the basis for the radioactive and hazardous waste management program are found in HNF-SD-SNF-RD-001 Spent Nuclear Fuel Project Standards/Requirements Identification Document. Specific requirements applicable to this chapter include:

- DOE Order 5400.5, Radiation Protection of the Public and the Environment
- DOE/RL-95-07 Hanford Site Air Operating Permit Application
9.3 RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT PROGRAM AND ORGANIZATION

9.3.1 Program Summary

The SNF Project waste management philosophy revolves around protecting workers, the public, and the environment, ensuring proper management of waste from its point of generation to its final disposition, and ensuring compliance with applicable federal, state, and local laws and regulations. To meet these goals, the SNF Project policy is to conduct operations and activities in a way that minimizes the quantity and toxicity of wastes generated, eliminates or minimizes pollutant releases to the environment, and minimizes the use of toxic substances.

SNF Project policy also requires that radioactive and hazardous material exposure of workers, visitors, the public, and the environment be as low as reasonably achievable. The waste management program is governed by the requirements of Section 16 of the SNF Project standards/requirements identification document (S/RID) (HNF-SD-SNF-RD-001). The following basic elements are included:

- Procedures governing solid waste treatment, segregation, monitoring, characterization, packaging, storage, transportation, and disposal.
- Waste minimization goals and practices that reduce generation of radioactive and hazardous waste, spread of contamination, and generation of nonhazardous refuse.

Hanford Site waste is handled by Waste Management Federal Services of Hanford, Incorporated (WMH), and the waste transportation organizations on a sitewide basis. The responsibilities and authorities of these organizations are presented in WMH-200 Series, Waste Management Operations Administrative Manual Procedures.

SNF Project waste management procedures are developed, prepared, reviewed, approved, issued, and maintained according to SNF Project administrative procedures. Procedures are in place for anticipated operations, maintenance, tests, and abnormal or emergency situations. Chapter 12.0 contains a description of the SNF Project facilities procedure program. At the SNF Project facilities, WMH uses a combination of SNF Project procedures and WMH internal procedures in the fulfillment of its waste management role.

The SNF Project follows the site waste minimization and pollution prevention management plan. Engineers and planners provide a checklist as part of that plan that states how they expect to minimize waste. For solid waste, a predetermination of waste characteristics is conducted to determine characteristics of materials that have been analyzed and to suggest alternatives to the use of hazardous materials. These tools aid in minimizing the generation of hazardous waste. Using these guides, introduction of asbestos or polychlorinated biphenyl-containing materials into the facilities is avoided.

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9.3.2 Waste Minimization Plan

Waste minimization is the prevention or reduction of waste at the source or reduction of the quantity of pollutants, contaminants, hazardous substances, or waste for treatment, storage, or disposal. Waste minimization is any source reduction or recycling activity that results in (1) reduction of the total volume of waste, (2) reduction of the toxicity of waste, or (3) both (1) and (2), as long as that reduction is consistent with the general goal of minimizing present and future threats to human health and the environment. The hierarchy of waste management practices places highest priority on source reduction followed by recycling, and finally treatment, storage, and disposal, in that order.

Section 16.7 of the SNF Project S/RID (HNF-SD-SNF-RD-001) describe waste reduction efforts that are applied at SNF Project facilities and are in agreement with the objectives described in the previous paragraph. The various levels of the waste minimization and pollution prevention programs are identified below. The Hanford Site waste minimization program is prescribed in HNF-PRO-462, Pollution Prevention, and is overseen by the Pollution Prevention Program Group. A Hanford Site waste minimization and pollution prevention awareness program plan targets all waste types — radioactive, hazardous, mixed, and nonregulated. The plan documents present and future activities that reduce the volume and toxicity of wastes. Waste minimization practices are integrated into operating procedures.

The SNF Project pollution prevention activities are implemented by facility administrative procedures and contain the key elements of source reduction, toxicity reduction, volume reduction, and product substitution. The work planning process evaluates waste generation activities for opportunities of pollution prevention. The program is based on HNF-PRO-462.

9.3.3 Radioactive and Hazardous Waste Management Organization

The SNF Project operations organization outlined in Chapter 17.0 administers the radioactive and hazardous waste management program within the SNF Project facilities. Facility personnel as waste generators, have the responsibility to ensure that all newly generated wastes are handled, stored, and packaged in accordance with requirements contained in Section 16 of the SNF Project S/RID (HNF-SD-SNF-RD-001). The radioactive and hazardous waste management processes are implemented by the SNF Project personnel identified in the following subsections.

9.3.3.1 Facility Manager: The facility manager at each SNF Project facility is responsible for daily functional activities and regulatory compliance associated with those activities (i.e., inspection, sampling, analysis, handling, packaging, storage, shipping, and disposing of wastes), ensuring compliance with employee training programs and approved procedures, ensuring adherence to approved plans, record keeping, reporting, and groundwater monitoring activities; supporting inspections by regulatory agencies responding to regulatory findings; and ensuring permitted activities are performed in compliance with permit requirements.
At the SNF Project facilities, WMH has contracted to provide waste management services including the following:

- Operation and maintenance of satellite accumulation areas, less-than-90-day pads, and low-level waste and transuranic management areas.
- Scheduling and removal of waste materials from satellite accumulation areas, less-than-90-day pads, and low-level waste and transuranic management areas to the appropriate treatment, storage, and disposal facility.
- Development and verification of waste documentation.
- Advice for planning waste generation and waste characterization, segregation, and packaging.

These contracted activities are performed in compliance with SNF Project procedures.

9332 **Environmental Compliance Officer**  An environmental compliance officer is assigned to assist the facility manager on waste management issues. The environmental compliance officer is responsible for regulatory compliance within the SNF Project.

9333 **Environmental Compliance and Support Services**  Environmental Compliance and Support Services performs activities associated with managing and permitting hazardous waste activities at SNF Project facilities, including developing the documentation required for managing treatment, storage, and disposal units and drafting permit applications to support operating activities.

9334 **Pollution Prevention**  The WMH Pollution Prevention Group serves as the lead Hanford Site coordinating group to support development and implementation of waste minimization and pollution prevention programs. This group, as described in HNF-PRO-462, will assist the facility manager in the development of successful programs, in the assessment of program effectiveness, and in preparation of regulatory required reports.

9335 **Hanford Analytical Services Management**  Hanford Analytical Services management provides guidance regarding the number of samples, sample types, sample quantities, and types of analyses required, reviews sample and analysis plans received from the waste owners for consistency with requests for laboratory services and capabilities, and provides data validation verification and coordination of laboratory support.

9336 **Facility Personnel**  Individual workers perform work activities in accordance with the specified work instructions, work procedures, and standard operating procedures. Each employee is responsible for promptly notifying management of events and conditions that could have adverse safety or environmental implications, including releases to the environment.
9.4 RADIOACTIVE AND HAZARDOUS WASTE STREAMS AND SOURCES

The following subsections identify SNF Project radioactive and hazardous waste management and handling processes. Facility-specific waste streams and sources are identified in the facility FSAR Annexes.

9.4.1 Waste Management Process

The goals and policies of the SNF Project hazardous and radiological waste management process are described in the following sections.

9.4.1.1 Solid Waste Management Process

The solid waste management process goals are as follows:

- Collect, store, and dispose of all waste in a safe, economical, and environmentally acceptable manner.
- Segregate, package, and ship nonradioactive wastes to the appropriate disposal or recycling facility.
- Segregate radioactive waste to prevent cross-contamination of radionuclides.

Criteria to ensure that the management process goals are met are specified in SNF Project control manuals and acceptance criteria documents. These internal documents meet the requirements specified in the SNF Project S/RID (HNF-SD-SNF-RD-001), and are in accordance with appropriate DOE orders, Hanford Site requirements, federal, state, and local regulations, and industry standards. Requirements for contact-handled solid wastes comply with the requirements of the appropriate SNF Project S/RID.

9.4.1.2 Air Quality Waste Management Process

Normal operations at the SNF Project facilities are not expected to generate any considerable amounts of air pollutants. Therefore, no air quality waste management process is necessary. The facilities are permitted, however, by the U.S. Environmental Protection Agency and Washington State Department of Health for air emissions.

Criteria for confinement, safe handling, and disposal of liquid and gaseous effluent streams are contained in Section 20 of the SNF Project S/RID (HNF-SD-SNF-RD-001).

9.4.1.3 Water Quality Waste Management Process

No unpermitted discharge of liquid or water pollutants to the environment is expected from the SNF Project facilities. The following normal precautions are necessary and will be followed by the SNF Project to protect the environment:

- Obtain permits and permit modifications for the SNF Project facilities, as applicable.
• Ensure septic systems are located, designed, installed, operated, maintained, and permitted according to WAC 173-272 (Sewage permits are issued by the Washington State Department of Health).

• Maintain facility operations within the authorization envelope.

• Ensure compliance with the requirements of the regulatory codes and applicable permits.

• Provide prompt notifications of failures in the sampling process or of any releases that exceed the requirements defined in applicable waste water permits required by Section 20.2 of the SNF Project S/RID (HNF-SD-SNF-RD-001).

• Ensure (where required for liquid discharges) that a facility effluent monitoring plan is issued and that annual reviews are provided in accordance with WHC-EP-0438, *A Guide for Preparing Hanford Facility Effluent Monitoring Plans*.

• Ensure training is provided for personnel who collect samples, verify and maintain personnel qualification records.

• Ensure proper use of sewage systems through strict adherence to operating procedures.

• Comply with spill prevention and reporting requirements as applicable.

9.4.2 Waste Sources and Characteristics

The following subsections summarize how and where SNF Project waste is generated and how it enters the appropriate waste handling or treatment system. Specific waste sources and characteristics are addressed in the facility FSAR Annexes.

9.4.2.1 Solid Waste Streams and Sources

The operations conducted by the SNF Project either directly or indirectly result in the generation of solid waste to be discarded. Solid waste is solid but may contain liquids or gases. It is sorted into solid waste streams on the basis of characteristics, constituents of interest, regulatory requirements (if applicable), and the waste acceptance criteria for final disposition. The following generally identifies the major sources and corresponding facility FSAR Annex. Criteria for the management of solid waste including its designation is provided in HNF-PRO-455, *Solid Waste Management* and meets the requirements of Section 16 of the SNF Project S/RID (HNF-SD-SNF-RD-001). The S/RID defines the steps in the waste designation process for mixed and hazardous wastes.

9.4.2.1.1 Radioactive Solid Wastes

Continued operations and decontamination and decommissioning of K Basins is expected to generate significant volumes of sludge, spent ion exchange modules, contaminated materials, and equipment. Only small amounts of hazardous or
Radioactive waste are expected to be generated within other SNF Project facilities. Solid waste could include sludge, spent water purification filters and units, contaminated equipment construction waste, compactable and noncompactable containerized waste from radiological control areas, decontamination waste used containers, and miscellaneous waste (i.e., waste generated by off-normal events such as spills or equipment failures and wastes generated during surveillance and monitoring operations). Most of these radioactive wastes will be designated as low-level waste, however, it is anticipated that the K Basins sludge will be remote-handled transuranic waste. Sources of radioactive solid waste include the following:

- **K Basins sludge** from storage of the SNF in the water basins and includes corrosion products from the storage racks and containers, dusts, corroded SNF, and other materials.

- **Water conditioning process components** including spent ion exchange modules and spent water treatment filters.

- **Low-level waste debris.** This waste stream consists of debris that is radiologically contaminated. The activities that produce this waste include the following:
  - Routine surveillance and maintenance activities. Tools and equipment used during routine surveillance and maintenance may create a waste stream.
  - Construction, demolition, and facility upgrades. Maintenance and removal of facility components and equipment generate a variety of low-level waste debris, including tools, wood, concrete, asphalt decontamination materials, and containment materials (e.g., tents and glovebags).
  - High-efficiency particulate air filter replacement (if they contain a sufficient level of activity present on the filters).
  - Decontamination. Fluids (water and nonhazardous decontamination fluid) and other materials are used in decontamination of equipment requiring maintenance and repair. Fluids are normally solidified by absorption in materials designed to meet landfill crush strength requirements.

### 9.4.2.1.2 Hazardous Wastes

Hazardous wastes produced by SNF Project operations consist primarily of maintenance waste (batteries and oily waste) and miscellaneous waste from the use of chemical cleaning agents or pesticides and herbicides. Hazardous waste will be recycled, as appropriate. There is a possibility that oils containing hazardous constituents will be used in the SNF Project facilities, resulting in any waste oil in this category being managed as a hazardous waste.
Oily debris is generated by maintenance and cleanup activities. This debris will be designated as hazardous waste based on the chemical constituents identified on the material safety data sheet or from analytical analysis. The source of this waste can be any of the following activities:

- **Routine operational activities**: Cleaning and degreasing of equipment systems produce oily and greasy rags and absorbents.

- **Routine maintenance activities**: Changing oil in compressor systems, crane hydraulics, and other permanently installed equipment creates a waste stream of hydraulic and engine oils and other lubricating fluids. These oils are absorbed in organic sorbents.

Identified hazardous solid wastes will be packaged and shipped to the appropriate regulated waste storage or disposal facility.

**9 4 2 1 3 Nonregulated Wastes** Nonregulated waste will be recycled as appropriate.

Nonregulated waste streams include the following:

- **Office waste**: This is disposed of by site contractors or through other approved means.

- **Vegetation growth around the facilities and animal carcasses (e.g., snakes, mice, ants, rabbits)**: These are managed and controlled by established site contractors. Removal when necessary is done in accordance with the applicable acceptance criteria for the waste type.

- **Sewage from the facilities**: This is disposed of via septic drain fields or collection tanks that are pumped and moved by established site contractors to approved facilities.

- **High-efficiency particulate air filters that are not classified as radioactive or hazardous waste**: These are disposed of by site contractors or through other approved means.

**9 4 2 2 Liquid Effluents and Sources** Radioactive water in the K Basins is treated to remove radioactive constituents by operation of water purification systems. Following the removal of the fuel from the basins, the water will be treated in site treatment effluent facilities.

Nonradioactive, nonhazardous liquid effluent streams are generated from routine operations such as hydrostatic testing, construction, and maintenance and from storm drains and condensate cooling coils. These streams discharge small amounts of water to the ground. Requirements for this type of effluent are contained in ST 4509, *Washington State Department of Ecology Waste Water Discharge Permit*. See the facility FSAR Annexes for specific effluents and sources.
9.4.2.3 Gaseous Effluents and Sources  Descriptive information for all the existing facility emission points and estimated emission quantities for all known emission points will be summarized in the update to DOE/RL-95-07. Passively ventilated, actively ventilated, and fugitive-type emissions will be addressed.

9.4.3 Waste Handling or Treatment Systems

Treatment methods for K Basins wastes are being evaluated, and treatment systems may be established for the sludge and basin wastes to allow them to be dispositioned. Such systems will be independently reviewed before they are implemented.

Nonregulated waste that can be unconditionally released is accumulated along with nonregulated waste generated outside the facility and stored in containers outside the facility.

Currently, no newly generated waste is treated by SNF Project operations because only existing wastes are being processed. The only "processes" applied to facility wastes are segregating, packaging, and transporting to a waste disposal facility. Details on the handling of each waste stream are contained in Sections 9.4 of the facility FSAR Annexes.

9.5 REFERENCES


WAC 173-272, 'On-Site Sewage Systems,' *Washington Administrative Code* as amended


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<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
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<td>FMEA</td>
<td>Failure modes and effects analysis</td>
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<td>FSAR</td>
<td>Final safety analysis report</td>
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<td>M&amp;TE</td>
<td>Measuring and test equipment</td>
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<tr>
<td>O&amp;M</td>
<td>Operations and maintenance</td>
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<td>ORR</td>
<td>Operational readiness review</td>
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<td>PAT</td>
<td>Preoperational acceptance test</td>
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<td>RAMA</td>
<td>Reliability, availability, and maintainability analysis</td>
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<td>SNF</td>
<td>Spent nuclear fuel</td>
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<tr>
<td>SSC</td>
<td>Structure, system, and component</td>
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<td>TSR</td>
<td>Technical safety requirement</td>
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10 0 INITIAL TESTING, IN-SERVICE SURVEILLANCE, AND MAINTENANCE

10 1 INTRODUCTION

Essential features of the initial testing program, the operational readiness review (ORR), the in-service surveillance program, and the maintenance program implemented at Spent Nuclear Fuel (SNF) Project facilities are described in this chapter. This chapter provides the basis for initial testing and maintenance activities through preoperational acceptance testing. This chapter describes the basis for preoperational testing, operational dry runs, the ORR, in-service surveillance, and operational phase maintenance. The initial testing program ensures that, at the time of initial operation, structures, systems, and components (SSCs) function as designed. The ORR verifies that the hardware, programs, and personnel are in place and effective in supporting safe operations of the SNF Project facilities. The in-service surveillance and maintenance programs help ensure that SSCs are available and perform as designed when needed. Discussions in this chapter demonstrate that a well-defined program with a commitment to testing, surveillance, and maintenance is an integral part of the overall safety assurance philosophy at SNF Project facilities.

The intent of this chapter is to accomplish the following objectives:

- Identify the requirements used to develop initial testing, ORR, in-service surveillance and maintenance programs.
- Describe the responsibilities of, and relationships among, the organizations having initial testing, ORR, in-service surveillance, and maintenance responsibilities.
- Present information on Hanford Site organizations that support initial testing, ORRs, in-service surveillance, and maintenance programs.

10 2 REQUIREMENTS

The requirements that form the basis for the initial testing program, the ORR program, the in-service surveillance program, and the maintenance program are found in HNF-SD-SNF-RD-001, Spent Nuclear Fuel Project Standards/Requirements Identification Document. Specific requirements applicable to this chapter include:

- DOE O 425 1, Startup and Restart of Nuclear Facilities
- DOE O 430 1, Life Cycle Asset Management
- DOE Order 4330 4B, Maintenance Management Program
- DOE Order 5480 19, Conduct of Operations Requirements for DOE Facilities
- RLID 425 1, Startup and Restart of Facilities

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10.3 INITIAL TESTING

The SNF Project initial testing program ensures the operability of equipment and facilities before facility operation. The initial testing program is developed and conducted in accordance with the requirements of DOE O 430 1 and includes testing of all SSCs to ensure compliance with design safety specifications and acceptance criteria.

10.3.1 Testing Program

The initial testing program consists of factory acceptance tests, construction acceptance tests, preoperational acceptance tests (PATs), and operational dry runs. SNF Project administrative procedures provide a planned systematic approach to testing activities and address the duties and responsibilities of participating organizations.

SSC testing requirements are extracted from applicable SNF Project documents, such as functional design criteria, Title II design documents (including architect-engineer generated test specifications), U.S. Department of Energy (DOE) orders and regulations, manufacturer testing and operational instructions, and this safety analysis report. These requirements, including any need for special testing, are compiled on a system by system basis into a system test specification. PATs are scheduled based on test sequence logic that takes into account system boundaries, system testing requirements, testing activity interfaces, and non-testing project activities such as critical path construction turnover and readiness review processes. Continuous online scheduling is used to facilitate response to design change-induced testing requirements and system and equipment availability. A punchlist process is established during startup testing to record and track incomplete work, test exceptions, outstanding approved design modifications, temporary modifications, and other open items on each system. A final punchlist is part of the release-to-operations package providing a comprehensive systems status in support of the ORR.

10.3.1.1 Factory and Construction Acceptance Tests

Factory acceptance tests are those tests performed in the factory that demonstrate compliance with construction and procurement specifications, or those operational tests that are best performed at the factory due to manufacturers' expertise or access restrictions on the field-installed equipment.

Construction acceptance tests are performed in the field by the construction organization to demonstrate compliance with installed requirements and to demonstrate quality of workmanship.

The factory and construction testing activities are monitored by construction management, the startup organization, the design authority, and DOE as required to ensure that the test requirements are met.

10.3.1.2 Preoperational Acceptance Tests

The PAT is a thorough test of the integrated system performance during normal and abnormal operating conditions. The PAT demonstrates that the system meets or exceeds the design requirements. The PAT is reviewed prior to testing by the test review board to ensure the test adequately demonstrates the test requirements. At the
completion of the PAT, a test summary report is developed summarizing the test results, identifying any problems encountered, and documenting the system status prior to turnover to operations. The test summary report and completed PAT are reviewed and approved by the test review board.

10.3.3 Operational Dry Runs  Operational dry run testing is performed to demonstrate the integration of the various SNF subprojects. Systems are brought online and operated under anticipated standard operating conditions using simulated special nuclear material and nonradioactive multi-cask overpacks. Dry run testing is performed by operations personnel with actual plant equipment and operating procedures. This testing demonstrates that operators and procedures are in a final satisfactory state of readiness to safely load and condition SNF, and transport receive, handle, and store multi-cask overpacks.

10.3.2 Test Procedures

Test procedures are developed, reviewed, and approved in accordance with the implementing instructions provided in the SNF Project startup administrative procedures. For PATs, the startup organization identifies test objectives, test methods, test conditions, applicable cautions, test boundaries, and acceptance criteria based on test specifications. The test procedure clearly describes test performance (e.g., initial conditions, sequence of testing, recovery actions, acceptance criteria). Test specifications, test procedures, and test reports are controlled, approved, and released in accordance with the implementing instructions provided in the SNF Project startup administrative procedures.

10.3.3 Review, Evaluation, and Approval of Test Results

Test results are reviewed to confirm that the results and the tests meet established requirements and that sufficient data have been obtained to proceed with further testing. A test review board facilitates technical review of test specifications, test procedures, test procedure changes, and test summary reports (including test results). The test review board consists of representatives from startup, operations, construction, design authority, safety, and quality assurance. Functions and responsibilities assigned to the test review board are defined by SNF Project administrative procedures. This board is responsible to ensure that nuclear and industrial safety considerations are incorporated into test documents and to ensure that critical facility and system characteristics are tested and meet defined acceptance criteria. The test review board recommends acceptance of test program results to SNF Project Operations management with the ultimate approval authority being the SNF Project director.

10.3.4 Test Program Control and Administration

The SNF Project startup administrative procedures provide the directions required to implement an effective test program, including the following:
• Testing control functions — instructions for conduct of testing, test deficiency reporting and resolution, punchlist tracking of open issues, and interfacing with engineering functions (design change initiation and control of modification-induced changes to the test program)

• Custody control functions — instructions for transfer of SSC custody from construction to startup and from startup to operations, and instructions for defining system test boundaries

SNF Project administrative procedures define the organization and responsibilities of the groups involved in the SNF Project facility startups. The SNF Project startup organization is responsible for startup management functions, including test program and sequence planning, test program implementation, punchlist item tracking and closeout, preoperational procedure preparation and review, test results review and approval, test specification development, and test records preparation. PATs are conducted by the SNF Project startup organization. The startup organization reports to the SNF Project operations organization and is independent of design and construction functions. The SNF Project startup organization manager has ultimate responsibility for establishing and implementing controls that ensure safety during execution of the startup program.

Test engineers are selected and trained in accordance with SNF Project startup administrative procedures. Operations and other support personnel are selected and trained in accordance with the program described in Chapter 12.0 of this document.

Records of the test program are kept using the guidance of HNF-PRO-210, *Records Management Program Standards*. These documents are retained as records in accordance with HNF-3552 *Spent Nuclear Fuel Project Execution Plan*.

### 10.3.5 Operational Readiness Reviews

The SNF Project facilities are classified as hazard category 2 facilities. These facilities require an ORR before restart of an existing facility or startup of a new nuclear facility in accordance with the requirements of Sections 1.7 of the standards/requirements identification documents (HNF-SD-SNF-RD-001) and RLID 4251. The purpose of the ORR process is to provide an adequate independent review of readiness to start or restart nuclear facility operations and to ensure that the facility or project can begin operation without undue risk to the public, onsite personnel, operating staff, or the environment.

The SNF Project implements the readiness review process described in RLID 4251 in accordance with the directions contained in SNF Project administrative procedures. Whenever a facility is started or restarted, guidance on approaches and methods approved as acceptable for implementing the requirements of RLID 4251 is provided in DOE-STD-3006-95, *Planning and Conduct of Operational Readiness Reviews (ORR)*. The SNF Project uses this guidance in the planning and implementation of ORRs.
An ORR verifies that the new facilities, facility modifications, new systems, new equipment installations, and associated operations are

- Operated safely
- Operated, maintained, and supported by trained and competent personnel
- Operated in conformance with applicable DOE orders and regulatory requirements
- Operated so that no undue risk to workers, the public, or the environment results
- Documented properly and adequately

The SNF Project facilities ORRs will be performed following a management self-assessment of readiness to start operations performed by the contractor. Responsible line management efforts to achieve readiness are conducted in accordance with the project management plan, startup plan, and other project management documents. The contractor validates readiness to proceed with the ORR. The operating contractor's ORR is followed by a DOE ORR. The Secretary of Energy has approval authority for initiation of radiological operations at the SNF Project facilities and for restart decisions following upgrades.

SNF Project facility ORR activities will be documented in formal ORR reports in accordance with Sections 1.7 of the standards/requirements identification documents (HNF-SD-SNF-RD-001) and with RLID 425.1. When only a small but manageable list of prestart findings remains, the approval authority prepares and forwards a readiness to proceed memorandum to DOE, Richland Operations Office. This action will result in the commencement of the DOE ORR.

10.4 IN-SERVICE SURVEILLANCE PROGRAM

The SNF Project in-service surveillance program is designed to maintain the integrity of facility systems and to ensure that systems perform their function of protecting the health and safety of the public, workers, and facility staff by prevention or mitigation of accident consequences. In-service surveillance of designated facility equipment and systems ensures technical safety requirement (TSR) performance requirements are met, as discussed in Chapters 5.0 of the facility annexes to the SNF Project Final Safety Analysis Report (FSAR). Chapters 5.0 of the facility FSAR Annexes explain how each TSR surveillance (test, calibration, or inspection of plant equipment) demonstrates operability or fulfills limiting condition for operations or limiting control setting. The analyses documented in Chapters 3.0 of the facility FSAR Annexes will define the numeric setpoints and values associated with each TSR.

The surveillance program is conducted in conjunction with the process standards and maintenance programs at the SNF Project facilities. SNF Project administrative procedures and the preface to the process standards describe the process of preparation, review, and implementation of process standards. Surveillance procedures providing in-the-field instructions for systematic inspection, testing, or calibration of plant equipment identified in the process standards are prepared using the guidance contained in SNF Project administrative procedures. Surveillance procedure performance is scheduled and tracked. SNF Project administrative
procedures describe the responsibilities for administering and using the tracking and scheduling process, including forecast reporting, overdue surveillance reporting, work package preparation, and generation of notices of discrepancies. Figure 10-1 is a flowchart illustrating the major steps in the performance of the surveillance testing process.

SNF Project administrative procedures define the actions to be taken when a failure to comply with TSR requirements is identified and pertinent action statements are successfully implemented. Tracking and resolution of this type of occurrence as a nonconformance report is described in HNF-PRO-052, *Corrective Action Management*, and SNF Project administrative procedures. SNF Project administrative procedures also define the actions to be taken when a violation of TSR requirements occurs. Response to this type of occurrence includes performance of the pertinent controls and responses specified in the administrative controls section of the TSR followed by occurrence tracking and resolution as specified in SNF Project administrative procedures.

The following sections describe the in-service surveillance program including provisions for testing and calibrations, control and calibration of measuring and test equipment (M&TE), trending and test result review, programmatic review, and training of surveillance personnel.

**10.4.1 In-Service Surveillance Measuring and Test Equipment**

The maintenance implementation plan for the SNF Project facilities discusses the control and calibration of M&TE for the maintenance program. This discussion also applies to M&TE used for in-service surveillance.

**10.4.2 Review and Trending of Surveillance Test Results**

SNF Project administrative procedures describe the requirement for, and the process used for, trending of surveillance test results. Cognizant engineers are responsible for surveillance test result review. Cognizant engineers use the Job Control System data processing database for the analysis of equipment history and system performance trends.

**10.4.3 Programmatic Review**

Inspections, audits, reviews, investigations, and self-assessments are necessary for an effective surveillance program. Title 10, *Code of Federal Regulations* Part 830, "Nuclear Safety Management," Section 830.120, "Quality Assurance Requirements" (10 CFR 830.120) requires management assessments to identify problems that hinder the organization from achieving its objectives. In addition, DOE Order 4330.4B requires that senior managers periodically review and assess elements of the maintenance program to assist line managers and supervisors in identifying and correcting program deficiencies. SNF Project administrative procedures define the management assessment process for SNF Project facilities. The procedure requires that the
The systematic approach used by the SNF Project to develop the overall maintenance plan includes the following elements:

- The design engineering organization specifically the design authority, prepares performance specifications that identify the need for reliability, availability, and maintainability or failure modes and effects analyses (FMEAs) for a system or component. This information is provided to the architect-engineer.

- The architect-engineer prepares the design addressing all elements/requirements of the performance specification. The result is drawings and procurement specifications.

- The architect-engineer prepares procurement specifications that include requirements for vendor submittals to address maintenance recommendations, spare parts recommendations, special tools or equipment recommendations, or any other required information for servicing, maintaining, or operating the system or component.
The architect-engineer also evaluates the need for reliability, availability, and maintainability analyses (RAMAs) or FMEAs and provides the master equipment list for the facility.

The vendor provides operations and maintenance (O&M) requirements/recommendations (often in the form of O&M manuals).

The design authority with the assistance of the architect-engineer evaluates the adequacy of the vendor's maintenance recommendations utilizing sound engineering judgment and considering the service conditions to be experienced by the equipment. The recommendations are also evaluated in light of the results of any RAMAs or FMEAs performed. The design authority then prepares the design baseline for the system or component, which includes the O&M manual, and provides this document to the operations organization which includes the initial testing function and the operations engineering function.

The design authority causes the O&M manual to be revised based on experience gained during initial testing.

The operation engineering organization specifically the assigned cognizant engineer, evaluates the O&M manual, prepares the maintenance requirements for the components and systems included in the master equipment list, prepares the maintenance procedures, and prepares the approved spare parts list.

The maintenance organization reviews, evaluates, and validates maintenance procedures recommends adjustments to the spare parts list based on maintenance experience and recommends adjustments to the maintenance requirements based on experience.

Facility management approves these plans and procedures.

During the design of the SNF Project subprojects, the architect-engineer uses hazard analysis or other analysis methods such as FMEAs to evaluate the need for features or actions to increase safety and reliability. These analyses identify equipment items that should be included in or removed from the design to simplify the operations or improve reliability. The hazard analysis process also identifies hazards resulting from identified maintenance activities. These hazards, if they result in accidents or contribute to an accident, will result in preventive or mitigative features or features that will decrease the frequency of the initial maintenance problem. In some cases where excess equipment capacity exists for short-term subprojects decisions are made to operate selected equipment to failure. For example, the results of these analyses for the Canister Storage Building did not identify maintenance activities that should have requirements similar to TSRs applied to them. In addition, the only Canister Storage Building TSRs related to maintenance identified in Chapter A5 0 of Annex A, the Canister Storage Building FSAR, and not associated with functional or calibration testing are those regarding changeout of high-efficiency particulate air filters when the filter load reaches a preset value.
A job analysis is performed to identify the scope of a job and associated critical tasks. Based on the results of the job analysis, training is developed as described in Section 12.2. This process, along with the design authority and management input, is the basis for determining specialized maintenance capabilities (e.g., vendor support or in-house journeyman knowledge) required. In addition, the SNF Project analyzes the types and complexity of equipment included in subproject design and, applying past experience with these types of equipment identifies the types and numbers of the various craftsmen needed for SNF Project maintenance support. To date, these analyses have resulted in the decision to initially have maintenance response teams assigned to each shift. The plans are to have the maintenance response teams be composed of a cross-section of craft and technical support including an engineer, a millwright, an electrician, and an instrument technician. This diversified team will have the capabilities to respond to problems identified by the facility operating staff, diagnose the problem, and to accomplish repairs. The remaining maintenance staff levels and levels of vendor support will be further defined based on a management review of the design and the specific maintenance requirements identified. Vendor and maintenance support staff will be available to perform maintenance that is beyond the capabilities of the shift response teams. Calibration and preventive maintenance tasks will be assigned to available qualified personnel, either shift response teams or support maintenance personnel based on work load. No maintenance tasks are planned for performance by operations personnel.

The following sections describe details of maintenance program implementation, including organization, training, facilities and equipment, post-maintenance testing, control and calibration of measuring equipment, trending, and limitations.

10.5.1 Maintenance Organization

DOE Order 4330.4B specifies that the organization and administration of the maintenance function ensure that a high level of performance in maintenance is achieved through effective implementation and control of maintenance activities. These activities include:

- Establishing written policies, procedures, and standards for maintenance
- Periodically observing and assessing performance
- Holding personnel accountable for their performance

Management policies and SNF Project procedures are in place to effectively manage SNF Project facility maintenance activities. SNF Project administrative procedures describe the philosophy, responsibilities, and processes for management of work activities.

The SNF Project organization including the maintenance and work control functions is described in Chapter 17.0. The SNF Project Operations Manager provides the overall management of O&M of each facility and provides consistent management direction for maintaining an effective work management system. The specific Facility Operations Manager is responsible for establishing a safe work environment while maintenance activities are being performed, maintaining consistent prioritization, adhering to an approved work schedule.
maintaining effective coordination among operations maintenance management, and support organization management to ensure full support of the work management systems and ensuring systems and equipment are acceptable after post-maintenance testing.

Managers of SNF Project support organizations or servicing organizations, including maintenance, are responsible for ensuring that the employees assigned to perform work in a facility use approved work documents that employees assigned to work are trained and qualified for the tasks to be performed, that work is only performed by employees when the activity is released by Operations, and for ensuring a safe work environment. The work management process for all categories of maintenance involves work screening, work completion, and retest, and post-work review. Each phase of the work management process considers factors related to health, safety, environment, and maintaining risks at a level as low as reasonably achievable. The process provides priority to work-affecting systems or equipment relied on for safety.

The maintenance work control manager is expected to provide or ensure that sufficient controls, training, and oversight are available to accomplish quality work, minimize rework, and avoid personnel-related problems such as procedure violations or bypassed quality control established hold points.

Procedures using a graded approach are provided for maintenance of equipment depending on the related design safety classification, the complexity of the work, and the potential hazard to personnel or equipment. Procedures for preventive, corrective, and predictive maintenance are prepared based on the guidance provided in SNF Project administrative procedures.

The work control administrative procedure assigns authority and responsibilities and defines the minimum responsibilities, requirements, and procedures used in performing maintenance, modifications, and fabrications. This procedure addresses the following:

- Personnel responsibilities for identifying deficiencies and initiating work requests that adequately describe symptoms or problems
- Supervisory responsibilities for controlling the conduct of maintenance activities and processing work requests
- Description of the process for initiating and processing work requests
- Definition of the priorities used to schedule work
- Determination of the impact of maintenance activities on facility operations
- Work planning and scheduling
- Requirements for personnel and equipment safety and for radiological protection
- Post-maintenance testing
• Collection of maintenance history for trending

The SNF Project facilities use a number of performance indicators to periodically observe and assess performance (e.g., open work packages, corrective maintenance completions, skin contaminations). Performance assessment activities are described in SNF Project administrative procedures.

10.5.2 Training of Maintenance Personnel

The SNF Project maintenance training program ensures that maintenance personnel possess the knowledge and skills necessary to perform their assigned duties in a safe, efficient, and cost-effective manner. The maintenance manager is responsible for determining the training and qualification needs of each individual in the maintenance organization. The maintenance manager and supervisors interface closely with the SNF Project training organization to establish and maintain course content, determine and support training schedules, implement on-the-job training, and provide feedback to adjust course content and emphasis.

Training and qualification of maintenance and work control personnel are defined in SNF Project administrative procedures. See Chapter 12.0 for a description of the training program.

10.5.3 Maintenance Facilities, Equipment, and Tools

DOE Order 4330 4B specifies that maintenance facilities, equipment, and tools should efficiently support facility maintenance and maintenance training. The SNF Project has evaluated the maintenance needs in this area and the necessary office areas, tool and equipment storage, equipment access, and utility services layout (e.g., instrument air, water, electric power) to support safe and effective maintenance are provided at the SNF Project facilities. Satellite work lay down and staging areas are located conveniently near maintenance work locations and take into account industrial safety requirements.

10.5.4 Post-Maintenance Testing

DOE Order 4330 4B specifies that post-maintenance testing requirements should be identified to verify that components will fulfill their functions when returned to service after maintenance. The post-maintenance testing requirements are identified, controlled, documented, reviewed, and accepted via the SNF Project work control process. Post-maintenance testing requirements are specified by the cognizant engineer in engineering-approved work packages and by the person-in-charge or manager as necessary for skill-of-the-craft work. The rigor of post-maintenance testing is applied on a graded approach, which is based on the extent of preventive or corrective maintenance performed and on the importance of the equipment to facility safety and reliability. These requirements are described in SNF Project administrative procedures.
10.5.5 Control and Calibration of Measuring and Test Equipment

DOE Order 4330 4B specifies that a program should be established for control and calibration of M&TE and that this program should ensure the accurate performance of facility instrumentation and equipment for testing, calibration, and repairs. M&TE includes all devices or systems used to inspect, test, calibrate, measure, or troubleshoot an instrument or piece of equipment to verify conformance to specified requirements. The calibration and test program is described in SNF Project administrative procedures and establishes requirements for the evaluation of out-of-calibration conditions identified during the periodic calibration of M&TE.

Each piece of M&TE is identified by a unique, permanently attached identification number. These numbers assist in identifying, tracing, and positively controlling M&TE. A master list of all M&TE provides the following identifying information:

- Generic description of equipment (trade or marketing name, manufacturer, model, and serial number)
- Unique identification number
- Calibration frequency
- Calibration due date
- Calibration procedure reference

Only M&TE that is calibrated to standards that are traceable to the National Bureau of Standards or other nationally recognized standards are authorized for use by SNF Project personnel. Instruments are calibrated at specified intervals, before and after use or just prior to use, as determined by required accuracy, intended use, frequency of use, stability characteristics, and other conditions affecting performance. When repair or calibration of M&TE is necessary, the recalibration must be traced to the National Bureau of Standards or to the standard of record for the M&TE. M&TE is controlled by the recall system of the Hanford Standards Laboratory. New M&TE or M&TE past its calibration date, in need of repair or with suspect calibration, is removed from the M&TE storage area and placed in a locked storage area until it can be sent to the Hanford Standards Laboratory or disposed of. Additional discussion of M&TE is included in Chapter 14.

10.5.6 Maintenance History and Trending

DOE Order 4330 4B specifies that a maintenance history and trending program should be maintained to document data, provide historical information for maintenance planning, and support maintenance and performance trending of facility systems and components. The SNF Project maintenance history and information system provides for this data retention service and allows for rapid retrieval of equipment identity and specification information and is maintained in...
accordance with SNF Project administrative procedures. The system provides a means to retrieve SSC maintenance histories as well as providing on a graded approach for trending types of maintenance performed to support analysis of problems.

10.5.7 Limitations on Maintenance

There are no known limitations at the SNF Project facilities imposed by design and operation on routine maintenance, renewal or repair of SSCs. Equipment in hazardous and radiological areas is designed to be free of maintenance as much as possible and is operated in a run-to-break mode. Accessibility for maintenance and provisions to support effective maintenance were considered in the facility designs.

10.6 REFERENCES


HNF-PRO-052, Corrective Action Management, Fluor Daniel Hanford Incorporated, Richland, Washington


HNF-SD-SNF-PLN-014 1999 *SNF Maintenance Implementation Plan (MIP)* Rev 0 Fluor
Daniel Hanford, Incorporated, Richland, Washington

RLID 425 1 1996 *Startup and Restart of Facilities*, U S Department of Energy, Richland
Operations Office Richland, Washington
Figure 10-1  Flowchart of Surveillance Actions
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CHAPTER 11 0

OPERATIONAL SAFETY
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LIST OF TERMS

DOE  U S Department of Energy
FSAR final safety analysis report
HFD Hanford Fire Department
SNF spent nuclear fuel
S/RID standards/requirements identification document
11 0 OPERATIONAL SAFETY

11 1 INTRODUCTION

Features of the Spent Nuclear Fuel (SNF) Project conduct of operations program and fire protection programs are described in the following sections and in the facility annexes to the SNF Project Final Safety Analysis Report (FSAR). The objective of this chapter is to provide a description of the essential features of the two programs as they relate to facility safety. Elements of other operational safety programs identified in Attachment 1 to DOE Order 5480 23, Nuclear Safety Analysis Reports, are covered elsewhere in this safety analysis report (e.g., fuel storage and criticality safety are addressed in Chapters 3 0 and 6 0, radiological protection is addressed in Chapter 7 0, hazardous material protection is addressed in Chapter 8 0, waste management is addressed in Chapter 9 0, and procedures and training are addressed in Chapter 12 0). The names and structures throughout the Hanford Site change frequently. Generic names (e.g., operations, engineering, facility management) are used in this chapter to demonstrate compliance with applicable requirements. See Chapter 17 0 for descriptions of organizations and institutional safety provisions.

11 2 REQUIREMENTS

The requirements that form the basis for conduct of operations and general aspects of operational safety are found in HNF-SD-SNF-RD-001, Spent Nuclear Fuel Project Standards/Requirements Identification Document, and the approved Conduct of Operations Graded Approach Applicability Matrix. Specific requirements applicable to this chapter include:

- DOE Order 5480 4, Environmental Protection Safety and Health Protection Standards
- DOE Order 5480 7A, Fire Protection
- DOE Order 5480 19, Conduct of Operations Requirements for DOE Facilities
- DOE Order 5480 20A, Personnel Selection Qualification and Training Requirements for DOE Nuclear Facilities
- Uniform Building Code (ICBO 1994)
11.3 CONDUCT OF OPERATIONS

"Conduct of operations" is a set of principles that establishes an overall philosophy for achieving excellence in the operation of the SNF Project facilities. The Project Hanford conduct of operations policy, which meets the applicable elements of DOE Order 5480 19, is documented in HNF-PRO-696, Conduct of Operations Policy. The degree to which each principle is applied is determined by the magnitude of the hazards and the complexity of the activities being performed by the facility staff.

Elements of the conduct of operations principles that are major contributors to safety performance at the SNF Project facilities are summarized in the following subsections or elsewhere in this safety analysis report (e.g., control of procedures and training is addressed in Chapter 12, and the operating contractor's process for handling events is discussed in Chapter 17.0). SNF Project administrative procedures implement the applicable guidance of DOE Order 5480 19. The approved conduct of operations graded approach applicability matrix for each SNF Project facility indicates the aspects of each principle that apply and the corresponding operating or administrative procedure that implements the principle.

11.3.1 Operations Organization and Administration

Excellence in operations is accomplished by establishing high standards by management, communicating operations standards (including worker safety standards) to the workers providing sufficient resources to the operations department, ensuring workers are well trained, monitoring operating performance, and holding workers and their managers accountable for their performance in conducting activities. SNF Project administrative procedures and HNF-3552, Spent Nuclear Fuel Project Execution Plan, outline the actions and responsibilities needed to implement a strong operations program. The organizational structure, responsibilities, and interfaces for the SNF Project are summarized in Chapter 17.0.

The facility operations manager is responsible for establishing specific goals and objectives for the operations organization and assigning responsibility for achieving these objectives. Performance measures are identified to assist in the measurement of success in meeting organizational objectives. Operating personnel are informed of the organizational objectives and their individual responsibility, authority, and accountability relative to these objectives. SNF Project administrative procedures describe the SNF Project goals identification process and address the issues of operational staff responsibilities, authorities, and accountability.

The operations personnel training requirements of DOE Order 5480 20A and DOE Order 5480 19 are implemented within the SNF Project facilities as described in...
SNF Project administrative procedures The shift manager certification program includes supervisor and management training in subjects such as leadership, interpersonal communications, motivation of personnel, and problem analysis and decision making.

Inspections, audits, reviews, investigations, and self-assessments are a part of the checks and balances needed in a successful operating program. Management performs routine observations of personnel performing operating activities. Deficiencies identified are documented, analyzed, trended, and corrected. In addition, other groups such as quality assurance also periodically review and assess operational performance. Programs to monitor personnel performing operating activities are described in SNF Project administrative procedures. Problem tracking, trending, and closure activities are implemented in compliance with SNF Project administrative procedures. Safety analysis report commitments are compiled in a commitment tracking log to facilitate report compliance.

11.3.2 Shift Routines and Operating Practices

SNF Project procedures define standards for professional conduct that ensure operator performance meets U.S. Department of Energy (DOE) and facility management expectations. The following subsections summarize the aspects of routine SNF Project shift activities and watch-standing practices that are important to safety. Chapter 17.0 discusses the integrated safety management system that denotes the philosophies, principles, and requirements of the Project Hanford Management Contractor's formal safety process.

11.3.2.1 Status Practices The operations staff manages, operates, and maintains the SNF Project facilities in a safe and efficient manner. Adherence to operating procedures and technical safety requirements helps to ensure that this objective is accomplished. Workers and their supervisors are held accountable for operating performance. Operators and operations shift management are notified promptly of all changes in facility status, operational abnormalities, or any difficulties operators encounter while performing assigned tasks.

11.3.2.2 Safety Practices As part of the conduct of operations program, operators must follow the requirements of the operating contractor's industrial safety program as described in the industrial health and safety manuals. The SNF Project hazardous materials program as described in Chapter 8.0 ensures that radiological, chemical, toxic material, and other exposure hazards are maintained as low as reasonably achievable. Strict adherence to procedures and posted personnel protection requirements ensure (1) appropriate use of monitoring instruments, (2) cognizance of permissible exposure levels, (3) proper use of and adherence to radiological work permits, and (4) effective and accurate deficiency reporting practices. Appropriate hearing, eye, head, foot, and respiratory protection is worn in designated areas to reduce the potential for illness and injury. Similarly, operators working at the facility exercise appropriate precautions when working with or around potential hazardous objects (e.g., ladders, electrical equipment, machines) or hazardous materials (e.g., chemicals, toxic materials) to reduce personal injury. Occupational injuries and illnesses are promptly reported. Radiological protection safety practices are addressed in Chapter 7.0. Personnel protective actions implement the requirements of the...
applicable standards designated in Section 19 of the facility standards/requirements identification document (S/RID) (HNF-SD-SNF-RD-001)

11 3 2 3 Operator Inspection Tours  Operators conduct periodic inspection tours of their areas of responsibility to ensure that the status of equipment is known. During a tour, equipment is inspected to ensure that it is operating properly or, in the case of standby equipment, that it is fully operable. The tour activities include, but are not limited to, inspection, troubleshooting, responding to alarms, housekeeping, logkeeping, and reporting deficiencies. SNF Project administrative procedures provide detailed direction to operations personnel on proper conduct of operator tours.

11 3 2 4 Tour Inspection Sheets  Tour inspection sheets (roundsheets and logs) are a means to uniformly record the status and condition of equipment and work areas. Use of operator roundsheets and logs is an effective method of providing operators with guidance on the extent to which equipment and areas are inspected. The sheets are also an effective place to record events and status of the areas and equipment inspected. Limits on the sheets remind operators of important control parameters for safe operation. The operator records notes on the sheets as a reminder of abnormalities and other items that should be reported to other operators or shift management. The roundsheets and logs are monitored periodically by shift management as part of normal shift routine to ensure that inspection tours are being conducted as required and that out-of-limit parameters are promptly corrected. SNF Project administrative procedures provide direction to operations personnel on the proper use of roundsheets and logs.

11 3 2 5 Response to Indications  Instrument readings are considered accurate and are responded to accordingly until inaccuracy is proven. Operators are trained to check other indications, if possible, when unexpected readings are observed. SNF Project administrative procedures provide direction to operations personnel on proper response to indications.

11 3 2 6 Resetting Protective Devices  When protective devices (e.g., circuit breakers, fuses) trip, an attempt is made to understand the cause of the trip before resetting. Before action is taken an evaluation will be performed to ensure that no unsafe or abnormal conditions exist that would preclude reset. SNF Project administrative procedures provide direction to operations personnel on proper precautions to be taken prior to resetting protective devices.

11 3 3 Control Area Activities

Control area activities are conducted in a manner that ensures safe and reliable facility operations. Operators are trained to be alert and attentive to indications and alarms. Indicators are monitored frequently and responses to alarms are swift, resulting in timely actions to correct alarm conditions. All reasonable actions are taken to clear alarming conditions. Distractions or ancillary duties that compromise an operator's primary responsibilities are minimized so that they do not interfere with the operator's ability to monitor and respond to facility parameters changes. Professional behavior is required at all times in designated control areas.
Control area conduct of operations principles will be applied in the SNF Project control rooms. Administrative procedures address the application of conduct of operations principles to the control areas.

### 11.3.4 Communications

Various communication devices are provided for transmission of information within the SNF Project (e.g., telephones, paging equipment, horns, bells, sirens, two-way radios). These devices are controlled to ensure that they do not detract from normal operations and are available in an emergency. The operating base for each shift position is equipped with adequate communication equipment to ensure proper performance of assigned duties. SNF Project administrative procedures delineate management expectations relative to formal, accurate, reliable and uniform face-to-face and remote communications during normal and emergency situations. Use of the public address system is administratively controlled to avoid excessive paging and unnecessary announcements.

Facilities and areas of the SNF Project are provided with systems (e.g., horns, bells, sirens) for communicating facility emergencies. If areas are identified where emergency systems cannot be heard, these areas are provided with alternative methods for alerting personnel including flashing lights, personal pagers that vibrate, or persons dedicated to make notifications. Emergency communication systems are tested periodically as part of a system surveillance procedure to ensure that they are functional.

### 11.3.5 Control of On-Shift Training

Operations training is controlled, and personnel being trained are carefully supervised to use the trainee's time effectively, to avoid mistakes during training operations, and to ensure that the trainee receives training within the job environment with as much hands-on experience as possible. SNF Project administrative procedures describe the on-the-job training program and controls. Operations personnel training has well-defined knowledge requirements and objectives for the trainee and is conducted under the direct supervision and observation of a qualified operator or instructor. The operator qualification program is based on small group instruction on the equipment being operated and on documentation of the training. Training is immediately suspended during unanticipated or abnormal events, accident conditions, or whenever the instructor believes suspension is necessary to ensure safe and reliable facility operation. The training program is described in Chapter 12.0.

### 11.3.6 Investigation of Abnormal Events

Chapter 17.0 describes the event investigation and lessons-learned programs used at the SNF Project facilities. These programs address the DOE Order 5480.19 guidance on investigating abnormal events including near miss situations.
11.3.7 Notifications

Chapter 15 describes the event notification program for the SNF Project facilities. This program addresses the DOE Order 5480 guidance on event notification to appropriate outside agencies. Instructions on internal notifications within the SNF Project are described in SNF Project administrative procedures.

11.3.8 Control of Equipment and System Status

Equipment and facility configuration is maintained within the design requirements through disciplined operation. Operations personnel are knowledgeable of operational limits and their responsibilities for actions that ensure compliance with these limits. Operators know the status of the equipment and operate systems using approved procedures. Direction on the use of procedures by facility personnel is addressed in SNF Project administrative procedures. Operations shift management is responsible for maintaining proper configuration and for authorizing status changes to major equipment and systems, which include all of the structures, systems, and components required to maintain the safety envelope as described in Chapters 40 of the facility FSAR Annexes. Shift management also ensures that operators of SNF Project equipment possess the necessary procedures, training, and qualifications. Nonroutine operations do not occur without specific approval by shift management. Operators are trained in accordance with approved procedures to take specific actions during emergencies to ensure the safety of personnel, the facility, and the environment without obtaining prior approval, however, the appropriate managers are promptly informed of these actions. Operators are trained that worker and facility safety should be achieved over facility production. Before first placing equipment or a system into operation, the individual components are checked for proper alignment and readiness for operation.

Equipment deficiency identification and documentation (e.g., tags, logs, and status boards) provide the necessary communication for removing equipment from active service until it is repaired, tested, and returned to service. The status of control panel and local panel alarms is readily available to operating personnel. Administrative controls include instructions for operators during installation of temporary equipment and when equipment is upgraded. These various forms of communication and administrative controls ensure that operators have the latest information to enable safe operation of the facilities.

11.3.9 Lockouts and Tagouts

The lockout/tagout program at the SNF Project facilities is established in accordance with DOE-RL-SOD-INST-L-T-001, Hanford Site Lockout Tagout Program. The lockout/tagout program is used to prevent personnel injury or equipment damage due to inadvertent activation during equipment operation, servicing, maintenance, or modification activities. SNF Project administrative procedures provide direction for SNF Project implementation of the lock and tag program requirements of the operating contractor's industrial safety procedures as required by...
29 CFR 1910, Section 1910.147, "The Control of Hazardous Energy" All SNF Project facility personnel receive fundamental lock and tag training. In addition, all authorized workers and all controlling organization-qualified workers receive detailed lock and tag training applicable to the functions that they perform.

11.3.10 Independent Verification

Independent verification is the act of (1) checking that a given operation conforms to established operational criteria, and (2) checking equipment component position independent of activities related to establishing the component's position. Operators are trained to perform independent verifications of component positions and to regularly consult the procedures and other reference materials that provide instructions on independent verification techniques. All components in systems that have safety-related functions are evaluated for the application of independent verification. Guidance on independent verification application is provided in administrative procedures, and each evolution requiring the application of independent verification is identified in the applicable operating procedure. At the SNF Project facilities, independent verification is applied to the lock and tag process. Double verification, as defined in Chapters 50 of the facility FSAR Annexes, is required for certain technical safety requirements.

11.3.11 Logkeeping

Narrative logs are established and maintained to provide an accurate history of SNF Project facility activities and to provide tools for reconstructing off-normal events. Logs are established for key operations shift positions in accordance with SNF Project administrative procedures. The log sheets provide accessible information and data associated with normal operation, testing, and off-normal activities for review by facility personnel.

11.3.12 Operations Turnover

Turnover guidelines are established and proceduralized to ensure that information required to adequately perform shift operations is documented by the shift going off and reviewed by the shift coming on. Hence, operations personnel will have an accurate picture of overall facility status. Personnel coming on shift review documentation such as daily operating roundsheets, logs, and checklists before assuming responsibility for their shift position. Shift management and operators going off shift are responsible for documenting equipment status, making entries on the roundsheets and logs, and apprising oncoming personnel of equipment status. SNF Project administrative procedures provide detailed direction on implementation of proper shift turnover practices and establish a turnover checklist to aid in effective communication of facility status at the time of turnover. Shift turnover practices at the SNF Project facilities include walkdowns of control panels by the appropriate operator and/or by shift management, and a shift briefing conducted by shift management coming on shift following turnover from the personnel going off.
The shift briefing ensures that operations and support personnel understand shift priorities and objectives.

**11.3.13 Operations Aspects of Facility Chemistry and Unique Processes**

Unique processes associated with evolutions at SNF Project facilities include hydrogen generation and maintenance of inerted atmospheres to prevent hydrogen fires. The hydrogen generation hazards are controlled via plans and procedures issued within the SNF Project. Administrative procedures provide guidance to operations personnel for control of unique processes. The procedures ensure that engineering (1) collects and evaluates data prior to a unique process operation and (2) establishes criteria that ensure operators are aware of parameter controls and recovery actions. Operators are qualified to survey and trend the required parameters, to recognize adverse conditions, to take appropriate action to provide timely reports of the condition to management, and to record required information.

**11.3.14 Required Reading**

The required reading program provides one method for disseminating various types of information applicable to SNF Project facilities to pertinent personnel. SNF Project administrative procedures provide direction for implementing this program. Types of documents applicable for required reading include selected procedure changes, selected occurrence reports, process standards and technical safety requirements, and selected training material. The SNF Project functional managers determine the appropriate material for the required reading list for their staff. The required reading program includes appropriate controls, including an acknowledgment sheet for the reader to indicate that the reading has been completed, and record retention measures.

**11.3.15 Timely Orders to Operators**

Timely orders allow management to rapidly disseminate essential daily or long-term directions and instructions to operating personnel to support operational activities. The timely orders contain concise information that is dated, prominently posted, and segregated into daily and long-term orders. A timely order is not used to change facility operating procedures but is incorporated into the appropriate procedure when the information is essential to facility operations. SNF Project administrative procedures provide direction for implementation of timely orders.
11.3.16 Operations Procedures

Chapter 12 describes the operations procedures program used by the SNF Project facilities. Chapter 12 describes the development process, content requirements, review and approval process, and requirements for the use of operations procedures. The process for procedure control of changes and revision is also described in that chapter.

11.3.17 Operator Aid Postings

An operator aid is a posting, diagram, sample schematic, or similar instruction intended to assist operators in performing their duties. A posting is an aid and therefore controlled by the process described in SNF Project administrative procedures if the following conditions apply:

- It is used by operators to perform their duties
- Changes to the information would affect the quality of the operator's work
- Postings are not controlled or required by other programs or information is taken from or referenced in an operations procedure

Operator aids are formal tools used to provide information to operators (but do not establish the facility operations baseline) and are, therefore, posted close to the area of expected use. Operator aids are approved by SNF Project management, are controlled in an operator aid log, and are periodically reviewed to ensure they are correct and necessary. Outdated aids are removed.

11.3.18 Equipment and Piping Labels

A standardized equipment labeling program at SNF Project facilities ensures that facility personnel are able to positively identify specific pieces of facility equipment. Label information meets regulatory requirements (e.g., Occupational Safety and Health Administration requirements) and is consistent with equipment descriptions used in facility procedures. All facility personnel are responsible for monitoring for missing or incorrect labels during performance of their normal functions or during assignment to housekeeping or facility condition inspections. SNF Project administrative procedures describe the labeling program, including specific direction on label information requirements, acceptable label material, proper label placement, and the categories of equipment that require labels.
11.4 FIRE PROTECTION

The SNF Project facilities' fire protection programs are implemented by SNF Project administrative procedures, which were developed in accordance with the criteria specified in Section 12.0 of the facility S/RID (HNF-SD-SNF-RD-001). SNF Project facility-specific elements of the fire protection programs are addressed in the facility FSAR Annexes.

Fire hazard analyses were prepared for all SNF Project facilities to meet the requirements of DOE Order 5480.7A. As required, these analyses addressed the following elements. Future changes to the fire hazard analysis will be screened to these elements to ensure that no unreviewed safety question is created.

- Description of construction
- Protection of essential safety-class equipment
- Fire protection features
- Description of fire hazards
- Life safety considerations
- Critical process equipment
- High value property
- Damage potential, maximum credible fire loss and maximum possible fire loss
- Fire department or brigade response
- Recovery potential
- Potential for toxic, biological, and/or radiological incident due to a fire
- Emergency planning
- Security considerations related to fire protection
- Natural hazards (earthquake, flood, wind) impact on fire safety
- Exposure fire potential, including the potential for fire spread between fire areas

11.4.1 Fire Hazards

Fire hazard analyses have been developed for the SNF Project facilities as required by Section 12.4.6 of the facility S/RID (HNF-SD-SNF-RD-001) and is implemented using HNF-PRO-340, *Fire Protection Program Overview*. A fire hazard analysis comprehensively assesses the risk from fire at the facilities to determine (1) whether the potential for the occurrence of fire is minimized, (2) that fire does not cause an onsite or offsite release of radioactive or other hazardous materials that will threaten the public health and safety or the environment, (3) whether requirements that will provide an acceptable degree of life safety to SNF Project personnel are in place and there are no undue hazards to the public from fire and its effects in SNF Project facilities, and (4) that safety systems are not damaged by fire.
11.4.2 Fire Protection Program and Organization

The fire protection program for the SNF Project facilities meets the requirements of Section 12.2 in the facility S/RID (HNF-SD-SNF-RD-001) and is structured and implemented in accordance with the operating contractor's safety management policies, philosophies and the criteria identified in SNF Project administrative procedures. The S/RID requires an administrative program that provides a level of fire protection that fulfills the requirements for the best protected class of industrial risks. The program is to be characterized by the inclusion of a continuing, sincere interest on the part of management and employees in minimizing losses from fire and related perils and the inclusion of preventive features necessary to ensure the satisfaction of objectives related to safety. Basic elements of the program must include the following:

- A reliable water supply for fire suppression
- Noncombustible facility construction
- A well-trained and equipped response force
- A means of notification of an existing fire
- Other protective measures identified in the hazard analysis

The SNF Project fire protection program (SNF Project administrative procedures) includes the following, as required by Chapter 12.0 of the facility S/RID (HNF-SD-SNF-RD-001):

- Minimize the potential for the occurrence of a fire or related peril by implementing industry standards and reduced risk criteria relative to fire protection and prevention (including National Fire Protection Association requirements)

- Ensure that fire does not cause an unacceptable onsite or offsite release of radioactive and other hazardous material that would threaten the public health and safety, worker safety, or the environment

- Establish requirements to provide an acceptable degree of life safety to SNF Project and subcontractor personnel and to ensure the public will not be exposed to undue hazards associated with fire and its effects in SNF Project facilities

- Ensure that process controls and safety systems are not damaged by fire or related perils

- Ensure that the SNF Project facilities will not suffer unacceptable delays as a result of fire and its effects

- Ensure that property damage from fire and related perils does not exceed an acceptable level

This policy applies to all operating contractor subcontractors and managed facilities, programs, projects, and activities.
Different aspects of the fire protection program are administered or addressed by various Project Hanford Management Contractor groups as follows:

- Fluor Daniel Hanford, Incorporated, Industrial Safety and Fire Protection, is responsible for policy development, program requirements, subcontractor overview, interpretations, and interfacing with DOE, Richland Operations Office.

- Hanford Fire Department (HFD) is responsible for maintaining a fully staffed, trained, and equipped response force for fire suppression, fire system inspection, testing, maintenance, and repair activities (except where interface agreements with other organizations exist), hazardous material, emergency rescue, and medical response for the Hanford Site, and administering the fire prevention program.

- Project Hanford Management Contractor Subcontractor's Fire Protection Engineers are responsible for program development, providing fire protection document reviews, providing technical assistance to the projects and facilities for implementing the fire protection program, performing fire protection facility assessments and fire hazard analyses. The Project Hanford Management Contract subcontractor is also responsible for establishing administrative programs that include fire prevention measures such as housekeeping practices, control of hot work activities, and storage of flammable materials, and employee training on fire prevention.

**11 4 2 1 Fire Protection Features** Facility-specific fire protection features are described in the facility FSAR Annexes.

**11 4 2 2 Spent Nuclear Project Fuel Fire Protection Program** The SNF Project fire protection program is managed by the SNF Project engineering organization. A full-time, qualified fire protection engineer is maintained on staff. The fire protection engineer is responsible for developing the fire protection program in accordance with DOE Order 5480.7A and Section 12 2.1 of the facility S/RID (HNF-SD-SNF-RD-001). The applicable requirements are implemented through SNF Project administrative procedures. A written agreement is maintained with the HFD Fire Maintenance Group for testing, inspection, and repair services. The HFD is responsible for fire suppression, emergency rescue, and medical response. The SNF Project fire protection engineer is responsible for performing fire protection document reviews, providing technical assistance to the SNF Project-related projects and facilities for fire protection program implementation, and performing fire protection facility assessments and fire hazard analyses.

Consistent with the facility S/RID (HNF-SD-SNF-RD-001), fire-related issues identified during the fire hazard analysis process (i.e., preliminary through final fire hazard analysis) were documented in an "open items" section of the preliminary fire hazard analysis.
11.4.3 Combustible Loading Control

Controls on flammable and combustible materials for all SNF Project facilities are implemented as described in HNF-PRO-359, *Control of Combustibles*, and SNF Project administrative procedures. These controls prevent or minimize the accumulation of flammable, combustible, and reactive materials. The essential requirements include housekeeping, control of hot work, control of flammable or combustible liquids, control of transient combustibles, control of construction sites, control of oxidizing and pyrophoric materials, proper storage of hazardous material, and disposal of waste absorbent material. See the facility FSAR Annexes for facility-specific combustible loading information. HNF-PRO-359 and HNF-PRO-076, *Safety Inspections*, describe the frequency and scope of housekeeping and safety inspections.

11.4.4 Fire Fighting Capabilities

The HFD maintains a training program for fire fighting, fire testing, and fire inspection. These training programs have been designed to meet National Fire Protection Association and DOE Order 5480-series criteria.

11.4.4.1 Fire Response Procedures The facility FSAR Annexes contain discussions of facility-specific fire response procedures.

11.4.4.2 Basic Training and Personnel Qualifications for Fire Fighters The HFD is responsible for training its employees, including response personnel, test and services fire fighters, and system maintenance craft personnel. The HFD also provides hands-on fire extinguisher use training for Hanford Site workers. Training records that record the levels achieved by individual HFD members using the National Fire Protection Association professional qualification standards are maintained. HFD members are trained to respond to events that occur in radiological and hazardous material environments.

11.4.4.3 Fire Fighting in Radiological and Hazardous Materials Environments

Radiological Environments WHC-IP-0939, *Hanford Fire Department Implementing Procedure*, establishes requirements and responsibilities for fire fighting and other emergency activities in radiological zones. The policy applies to all members of the HFD and addresses emergency operations and training. The policy identifies personnel responsibilities, precautions for wild land fires involving radiological zones, and requirements for protective equipment, dosimetry, and emergency dose limits.

Hazardous Material Environments A hazardous materials emergency response plan and a hazardous materials operations plan are maintained by the HFD. The emergency response plan is a general outline for all hazardous material responses designed to meet the requirements of NFP A 472, *Standard for Professional Competence of Responders to Hazardous Materials Incidents*. The operations plan is a detailed step-by-step guideline that includes specific actions to be taken during an incident.
11 4 5 Fire Fighting Readiness Assurance

11 4 5 1 Fire Prevention The Hanford Site fire marshall is responsible for preparing and managing a fire prevention program in accordance with the requirements of NFPA 101, *Life Safety Code*. The Hanford Site fire prevention program includes annual facility employee training on recognizing and controlling fire hazards, on response and notifications in the event of fire, and on use of portable fire extinguishers. This program also includes a requirement for annual facility inspections, performed by the HFD, to identify potential fire hazards, to verify reliability of fire protection systems, and to verify emergency access for response personnel. Written inspection findings are provided to the facility manager, who is assigned responsibility for identifying and implementing corrective actions.

11 4 5 2 Types and Frequencies of Fire Safety Drills and Exercises In association with the Hanford Site emergency preparedness organization, the HFD participates in site emergency exercises. These exercises are managed by the site emergency preparedness organization and are scheduled on a monthly basis. Field exercises requiring full HFD participation are typically conducted on a quarterly basis. Desktop exercises requiring HFD participation in the form of representation (typically by an incident commander) are generally conducted on an annual basis. The HFD may be requested by the SNF Project facility manager to participate in facility fire drills. Facility fire drills are scheduled and conducted by the emergency preparedness organization as part of the annual drill schedule. Drills are performed to satisfy the requirements of DOE O 151 1, *Comprehensive Emergency Management System*, which requires the facility to conduct a range of drills that represent credible events that can occur at the facility. The goal of the fire drill program is to involve each shift in at least one fire drill per year.

11 4 5 3 Fire Protection Record Keeping The Hanford Site fire marshall maintains records on the following items:

- General examinations and approvals conducted by the Hanford Site fire marshall
- Fire prevention inspections and findings
- "Permitted areas or operations"
- Fire investigations
- Fires
- Hanford Site project reviews in support of the Site Selection Review Board

Records of all fire protection system acceptance testing are maintained by the HFD manager, Fire Protection Systems Administration and Testing. Other fire protection records (i.e., fire extinguisher inspections, fire barrier inspections) are maintained by facility management.

11 5 REFERENCES


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DOE Order 5480 4, Environmental Protection, Safety and Health Protection Standards, U S Department of Energy, Washington, D C


DOE Order 5480 19, Conduct of Operations Requirements for DOE Facilities, U S Department of Energy, Washington, D C

DOE Order 5480 20A, Personnel Selection, Qualification and Training Requirements for DOE Nuclear Facilities, U S Department of Energy, Washington, D C

DOE Order 5480 23, Nuclear Safety Analysis Reports, U S Department of Energy, Washington, D C

DOE Order 6430 1A, General Design Criteria U S Department of Energy, Washington, D C


HNF-PRO-076, Safety Inspections, Fluor Daniel Hanford, Incorporated, Richland, Washington


HNF-PRO-359, Control of Combustibles, Fluor Daniel Hanford, Incorporated, Richland, Washington


ICBO, 1994, Uniform Building Code, International Conference of Building Officials, Whittier, California


CHAPTER 12 0

PROCEDURES AND TRAINING
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<td>DOE</td>
<td>U.S. Department of Energy</td>
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<tr>
<td>FSAR</td>
<td>Final safety analysis report</td>
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<tr>
<td>SNF</td>
<td>Spent nuclear fuel</td>
</tr>
<tr>
<td>S/RID</td>
<td>Standards/requirements identification document</td>
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</table>
12.0 PROCEDURES AND TRAINING

12.1 INTRODUCTION

This chapter describes the essential features of the Spent Nuclear Fuel (SNF) Project procedures and training programs. These programs are designed to ensure that personnel are well qualified and prepared to perform their duties in a safe, effective, efficient and environmentally sound manner. SNF Project management recognizes the importance of effective procedures and personnel training in achieving the highest level of safety in facilities operation and maintenance. Line managers are responsible for staffing levels that support completion of required training.

Safety management policies used as a basis for the procedures and training programs are contained in HNF-3552 *Spent Nuclear Fuel Project Execution Plan*. Processes used to meet technical procedures and training program safety requirements are documented in SNF Project administrative procedures.

A structured process for developing, maintaining, and delivering procedures and training ensures that the safety hazard and accident analyses summarized in Chapters 3.0 of the facility annexes to the SNF Project Final Safety Analysis Report (FSAR) form the basis of the technical content of operating procedures and training for normal, abnormal, and emergency conditions. This process also ensures that specific procedures and training, as described in other chapters of this document, are systematically developed and maintained to meet the regulatory requirements for each safety-related topic.

- Prevention of inadvertent criticality (Chapter 6.0)
- Radiation protection (Chapter 7.0)
- Hazardous material protection (Chapter 8.0)
- Radioactive and hazardous waste management (Chapter 9.0)
- Initial testing, operational readiness reviews, in-service surveillance and maintenance (Chapter 10.0)
- Conduct of operations and fire protection (Chapter 11.0)
- Quality assurance (Chapter 14.0)
- Emergency preparedness (Chapter 15.0)
12.2 REQUIREMENTS

The requirements that form the basis for the procedures and training programs are found in HNF-SD-SNF-RD-001, *Spent Nuclear Fuel Project Standards/Requirements Identification Document*. The specific requirements applicable to this chapter include:

- DOE Order 5480 19, *Conduct of Operations Requirements for DOE Facilities*
- DOE Order 5480 20A, *Personnel Selection Qualification Training and Staffing Requirements at DOE Reactor and Non-Reactor Facilities*

12.3 PROCEDURE PROGRAM

SNF Project activities are conducted in accordance with written procedures. SNF Project Training and Support Services specifically the procedures organization, develops, maintains, and controls procedures in conformance with applicable federal and state regulations U S Department of Energy (DOE) orders, and industry standards and codes. HNF-PRO-552, *Project Controls*, provides guidance on the responsibilities, requirements, and approaches for controlling administration, baseline, design, construction, reporting, and closeout activities for Hanford Site projects. It references additional Site manuals that provide specific requirements and approaches applicable to a project including quality assurance program requirements. Existing Site procedures used as applicable, on all Hanford Site projects represent the procedure program for SNF Project design, construction, and procurement activities. The SNF Project quality assurance program plan and implementing procedures control SNF Project work processes.

Technical procedure development and control processes are based on the guidance of HNF-PRO-229, *Technical Procedure Standard*, which defines minimum requirements for facility and organization technical procedure development and use. HNF-PRO-229 establishes processes for the identification of need, preparation, review, approval, change, revision, and periodic review of procedures for production, testing, operations surveillance, and maintenance activities.

Procedure users use and comply with approved technical procedures as required by an assigned classification code (except when an employee received direct management or supervisory approval during emergency or off-normal conditions). SNF Project administrative procedure instructions for using and complying with procedures are different depending on the classification code. Classification descriptions for technical procedures include:

- Step by step — the nature of the task requires this type of rigor because (1) potential difficulties are present either through the complexity of the procedure, the nature of the work or the task affecting other components within a more sensitive system or (2) a safety problem or damage to the equipment could occur if this procedure is not followed correctly.
General intent — those procedures where the task can be accomplished using the skill of the user, indicates that the task is performed routinely and indicates there is little to no liability to personal safety, economic value, the environment, or equipment failure if performed out of sequence.

HNF-PRO-589, *Processing Project Hanford Procedures*, states that all managers are responsible for work performed in their divisions and for ensuring the work is accomplished in accordance with established procedures. Specific responsibilities and requirements associated with procedure use and compliance are established in SNF Project administrative procedures. The SNF Project procedures organization is responsible for promptly providing operations with the most recent version of all applicable procedures. It is the managers' responsibility to supply controlled copies of procedures and instructions at work locations. The procedure user has the responsibility to ensure that the procedure to be used is the most current by verifying the hard copy against the electronic copy or index on the controlled file server, or, if the electronic version is unavailable, by checking it against the backup hard copy or by contacting the procedures organization.

### 12.3.1 Development of Procedures

Section 1.2 of the standards/requirements identification document (S/RID) (HNF-SD-SNF-RD-001) requires that procedures be developed for all anticipated operations, evaluations, tests, and abnormal or emergency situations. The extent of detail in a procedure depends on the complexity of the task, the experience and training of the user(s), the frequency of performance, and the significance of the consequences of error. Section 1.2 of the S/RID further requires that the methods for developing procedures are clearly defined, including descriptions of procedure formats. SNF Project administrative documents provide guidance for standards and requirements identification document requirements, personnel responsibilities, procedure development, procedure change, and procedure use and compliance. All technical procedures are prepared and revised in accordance with applicable SNF Project administrative procedures. SNF Project administrative procedures delineate the process and requirements for the preparation or modification of administrative procedures.

The SNF Project technical procedures program is controlled and applied using SNF Project administrative procedures. These controlling procedures meet the applicable requirements of DOE Order 5480 19. Startup administrative procedures and startup technical procedures are prepared and revised in accordance with applicable SNF Project administrative procedures.

Primary steps in the technical procedure development process are illustrated in Figure 12-1 and are described in the following subsections.

#### 12.3.1.1 Identify the Need

Technical procedures are developed for anticipated operations: transients, evolutions, surveillances, maintenance, and abnormal or emergency situations.
A new or revised procedure may be identified when modifications in the conduct of an operation are implemented, when equipment or systems are modified, when a procedure is deemed inadequate during task performance, or as a result of a periodic review of technical procedures.

12.3.1.2 Develop the Technical Basis  During the technical draft development phase, a subject matter expert gathers information that will lead to identifying the sequence of steps that should be performed in a particular process (i.e., the technical basis for the procedure). A Technical Basis for Procedure Development form documents the technical basis. Source documents used in developing the technical basis for a procedure include, but are not limited to, the following:

- Safety analysis report
- Technical safety requirements
- Safety evaluation report
- Standards and requirements identification documents
- Process standards
- System design descriptions
- Facility configuration
- Vendor information
- Operational lessons learned
- Functions and requirements documents

12.3.1.3 Prepare and Review the Draft  Draft procedures are prepared consistent with administrative procedure requirements. A writer's guide is used to ensure the format and content of each procedure is consistent, the procedure steps are written to effectively communicate the required operator actions, and the procedure incorporates human factors that lead to effective procedure use.

Technical review and verification ensures the technical accuracy of a procedure, ensures the procedure incorporates human factors principles, and compares the procedure against the appropriate source document requirements (e.g., system design descriptions, functions and requirements documents, DOE orders, technical requirements, regulatory requirements).

Technical procedure validation is a review of a procedure performed by the end user to ensure its usability and correctness. This validates that the procedure provides sufficient and understandable guidance and direction to the user and that the procedure is compatible with the equipment or system being maintained.

12.3.1.4 Review and Approve the Procedure  New procedures, procedure revisions, and technical changes to procedures are reviewed and approved according to approval designator requirements contained in SNF Project administrative procedures. Approval designators verify that environmental, safety, health, and quality assurance requirements have been properly addressed. Document approval is indicated by a signature from the functional reviewer.
12.3.1.5 Release and Use the Procedure  All SNF Project administrative and technical procedures are assigned a procedure and revision number. A record copy is placed into the procedure master file, and working and controlled copies of the procedures are made available through electronic media.

12.3.2 Maintenance of Procedures

The safety of SNF Project facilities and personnel depends on the availability of operating, maintenance, and alarm response procedures that correspond to the current plant configuration. To ensure that personnel use only the most current procedures, the SNF Project has implemented a process that provides timely review, approval, change, revision, and control of procedures consistent with the requirements contained in Section 1 2 3 of the S/RID (HNF-SD-SNF-RD-001).

This process for technical procedure maintenance and review is documented in SNF Project administrative procedures. Training needs related to procedure revisions is determined by the line manager who is responsible for the procedure being revised. This determination is based upon the significance of the change. For changes that require new skills or knowledge, the line organization would request the training department to assist in determining a training need. For changes that do not involve new skills or knowledge but need to be communicated, the line organization would use the required reading process. Employees are required to read revisions that apply to their duties prior to implementation of the revisions and are required to document this training per SNF Project administrative procedures. For changes that are not deemed significant or are editorial, the line organization would denote no action.

12.3.2.1 Procedure Change  Section 1 2 of the S/RID (HNF-SD-SNF-RD-001) requires that procedures be technically and administratively accurate and that they incorporate appropriate facility design, safety analysis, and vendor technical information. To satisfy these requirements, technical and administrative procedure changes are made in accordance with the guidance of HNF-PRO-229 and SNF Project administrative procedures. Changes to Project Hanford Procedures are controlled by the process described in HNF-PRO-589, which requires a review and screening of the changes, including a screening for facility impact, by appropriate Project Hanford Management Contract contractor personnel.

The procedure change process is used to proceduralize modifications to essential equipment, processes or requirements, and to correct procedural errors and ambiguities and/or human factor deficiencies that could result in operator error or unsafe job performance. Procedure modifications also result from issues identified during training activities and from efforts to resolve operating occurrences. SNF Project administrative procedures require that procedure users stop work if the work cannot be accomplished as described in the procedure or if accomplishment of the work would result in an undesirable situation. Under these circumstances, the procedure user is required, by procedure, to notify supervision of the problems.
The level of review and approval for procedure changes depends on the scope of the recommended change, and the approval process is addressed in SNF Project administrative procedures.

12 3 2 2 Periodic Review of Procedures Section 1 2 2 of the S/RID (HNF-SD-SNF-RD-001) requires that procedures be reviewed at periodic intervals to ensure that information and instructions are technically accurate and that appropriate human-factor considerations have been included. SNF Project administrative procedures implement a periodic review process that ensures the technical accuracy and the proper consideration of human factor issues in procedures that implement the safety basis. This process specifies that procedures affecting the safety-class and safety-significant systems are reviewed at least every two years. Administrative procedures that implement the requirements identified in technical safety requirement administrative controls are also reviewed at least every three years.

The due date for a periodic review is defined as two years (one year for emergency response procedures) after the previous revision issue date. The procedures program organization maintains a recall system that will initiate the periodic review process by notifying the designated technical authority of the due date for a review. The technical authority is responsible for ensuring that the procedure is technically adequate, accurate, meets current facility design, and meets SNF Project content and format requirements. The technical authority also identifies other reviewers who should participate in the review process for the particular procedure undergoing periodic review. All reviewers are required by procedure to be trained to perform their assigned review assignments. The SNF Project administrative procedure addressing the periodic review process includes a table of positions that are designated as reviewers or approvers of the various types of Project procedures.

12 4 TRAINING PROGRAM

Section 4 1 1 of the S/RID (HNF-SD-SNF-RD-001) requires that a nuclear facility operating organization have an initial and continuing training program to ensure that operating organization personnel are qualified to perform job requirements. This program is to be achieved by using a systematic approach to training. The training program for the SNF Project operating organization is consistent with these requirements.

The objective of the personnel training program is to provide and maintain a qualified work force for safe and efficient facility operations. Training is one element of an employee's qualification which typically includes a combination of education, experience, and job-specific training. Managers identify training and qualification requirements for their personnel from federal and state regulations, DOE orders, industry standards, company manuals, and personnel performance. Individual training must be current before performance of tasks requiring training.
Selection and qualification requirements are established for operators, training instructors, and maintenance staff and technical staff positions. Operations organization management is responsible for establishing selection and qualification requirements. Nuclear chemical operator and shift manager positions require certification as described in DOE Order 5480.20A.

The SNF Project procedures and training organization develops classroom, self-study, and on-the-job training programs consistent with the requirements contained in DOE Order 5480.20A. The sitewide training organization offers courses that meet general training requirements in the program areas of general employee orientation, health, safety, and environmental and occupational radiation protection. SNF Project Training and Support Services, specifically the Training organization, provides training support to ensure that facility- and job-specific training requirements are satisfied.

12.4.1 Development of Training

Section 4.1.1 of the S/RID (HNF-SD-SNF-RD-001) requires that training design, development, and implementation be based on learning objectives. The S/RID also requires that material, such as lesson plans and on-the-job training guides, training aids, and student materials, are developed to conduct training. A graded approach is allowed to be used to develop training material. Trainers are required by Section 4.2.1 of the S/RID to be qualified both in the subject being presented and in instructional skills.

The SNF Project training program is developed and implemented through a joint effort between the SNF Project Training and Support Services organization and line organizations (i.e., Operations, Maintenance, Radiological Control). SNF Project management maintains full responsibility for ensuring that plant training needs are satisfied. All formal qualification training is provided by qualified instructors or experienced vendors.

The training program applies performance-based training to specific job requirements including factors such as the following:

- Plant operations and design
- Instrumentation and control
- Methods of dealing with process malfunctions
- Emergency procedures
- Radiological control
- Hazard communication
- Hazardous material recognition
- Criticality

SNF Project Training and Support Services, specifically the Training organization, applies a graded systematic approach to training development, as indicated in Section 4.2 of the S/RID.
(HNF-SD-SNF-RD-001) This systematic approach, as it is applied within the SNF Project, is developed and documented in SNF Project administrative procedures. This performance-based approach to training comprises five phases: analysis, design, development, implementation, and evaluation.

12.4.2 Maintenance of Training

The SNF Project recognizes that training programs will only be effective if they reflect current operating practices, conditions, and procedures. To ensure that training properly reflects operating practices and procedures, SNF Project administrative procedures implement a training materials maintenance process that is used to track items that may affect the content of SNF Project training programs and materials. This process is accomplished in conjunction with the configuration management process and permits the training organization to respond to requirements for changes resulting from revised regulatory requirements, safety analyses, technical safety requirements, procedural changes, changes in the facility equipment configuration, feedback from course evaluations, feedback from operating experience, and resolution of audit findings. The training materials maintenance system is used to document the origin of the change request, the evaluation and disposition of the change request, and the recommended changes in program materials. Training content and materials are revised using the same process that is used to develop new training materials.

The effectiveness of the training program is maintained through feedback mechanisms such as course evaluations and training program evaluations. Competency of the instructional staff who deliver the facility training is maintained through compliance with initial and continuing instructor qualification requirements contained in SNF Project administrative procedures.

SNF Project training records are managed in accordance with SNF Project administrative procedures. Official training records are maintained in the central Training Services organization in accordance with the requirements of DOE Order 5480.20A. Courses completed, applicable training expiration dates, and summary reports are available electronically to managers and designated individuals.

12.4.3 Modification of Training Materials

The need to modify training materials may be identified through the configuration management system as a training deficiency identified by students, instructors, operations or maintenance staff, or management. In addition, training packages will be reviewed after any significant program change and will be updated in accordance with the requirements of DOE Order 5480.20A to ensure that information in the training package is current with plant operations. Per SNF Project administrative procedures, all changes to training program content, together with the reason for the changes.
12.5 REFERENCES


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Figure 12-1 Procedure Development Process

1. Identify need
2. Develop technical basis
3. Write draft
4. Review draft
   - Verification
   - Validation
5. Resolve comments
6. Approve procedure
7. Release and use

Requests and records:
- Working copy
- Records
CHAPTER 13 0

HUMAN FACTORS
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<th>Acronym</th>
<th>Description</th>
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<tbody>
<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
</tr>
<tr>
<td>DRD</td>
<td>Design requirements document</td>
</tr>
<tr>
<td>FSAR</td>
<td>Final safety analysis report</td>
</tr>
<tr>
<td>HFE</td>
<td>Human factors engineering</td>
</tr>
<tr>
<td>HMI</td>
<td>Human–machine interface</td>
</tr>
<tr>
<td>RFP</td>
<td>Request for proposal</td>
</tr>
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<td>SNF</td>
<td>Spent nuclear fuel</td>
</tr>
<tr>
<td>S/RID</td>
<td>Standards/requirements identification document</td>
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<tr>
<td>SSC</td>
<td>Structure, system, and component</td>
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13 0 HUMAN FACTORS

13 1 INTRODUCTION

Human factors engineering (HFE) is important in assuring the safe operation of the Spent Nuclear Fuel (SNF) Project facilities. This chapter identifies appropriate HFE requirements and describes the process to incorporate HFE into SNF Project facilities. The facility annexes to the SNF Project Final Safety Analysis Report (FSAR) contain an HFE review of the significant facility structures, systems, and components (SSCs) that include human–machine interfaces (HMIs), a review of abnormal sequences involving human operators, and a discussion of how the SNF Project human factors program will be used for design modifications and operation of SNF Project facilities.

HFE considerations are commensurate with the following:

- Planned SNF Project mission
- Hazard category 2 classification of SNF Project facilities
- Level of complexity of the SNF Project operations
- Phase of the system life cycle for which U S Department of Energy (DOE) approval is sought
- Potential consequences of a human error and the likelihood of error occurrence
- Type of the human operations with the SSCs and the SNF Project processes

HFE is the application of knowledge about human performance capabilities and behavioral principles to the design, operation, and maintenance of human–machine systems so that personnel can function at their optimum level of performance.

The incorporation of HFE considerations into the SNF Project facilities designs and operational processes ensures that human contributions to operational success are enhanced while abnormal situations resulting from human error are reduced. Facility design and operations that include insights from a human factors perspective help to ensure that facility and equipment design, staff capabilities and operational procedures all combine to enhance human performance while protecting against susceptibility to human error.

Staffing and qualifications of personnel, personnel training and written procedures to guide operations and maintenance are covered primarily in Chapters 10 0, 12 0, and 17 0. Allocation of human–machine functions were considered by the design team using structured and systematic methods. A design goal was to reduce human operation to a reasonable degree and move systems into a more automatic function. Human involvement in this case is delegated to routine.

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maintenance of these automatic systems, and thus reduces both the complexity and interaction of human to system. An example of allocation function is the design team looking at the weights and materials and determining that machinery is needed for handling large weight equipment that utilizes handling devices like the receiving crane. Human–machine allocations were also appropriately considered during the design of the multi-camister overpack handling machine. Additional human factors concerns are addressed as described in this chapter.

The specific facility FSAR Annexes discuss the HFE design process as implemented in the facility.

See Figure 13-1 for inclusion of the overall method for human factors into the design of the system.

13.2 REQUIREMENTS

According to DOE Order 5480 23, Nuclear Safety Analysis Reports, Attachment 1, Section 3 c (1)(b), human factors safety includes allocation of control functions to humans and machines, staffing and qualifications of personnel, personnel training, written procedures to guide operations and maintenance, and design of HMI’s. Human factors design activities focusing on each of these areas responding to governing DOE orders and standards, and non-DOE referenced standards, commensurate with classification of the SNF Project facilities as category 2 nonreactor nuclear facilities. Specific requirements applicable to this chapter include:

- DOE Order 5480 19, Conduct of Operations Requirements for DOE Facilities
- DOE Order 5480 20A, Personnel Selection Qualification and Training Requirements for DOE Nuclear Facilities
- DOE Order 5480 23, Nuclear Safety Analysis Reports
- DOE Order 6430 1A, General Design Criteria, Section 1300-12
IEEE 1023-88, Guide for the Application of Human Factors Engineering to Systems Equipment and Facilities of Nuclear Generating Stations

HNF-SD-SNF-RD-001, Spent Nuclear Fuel Project Standards/Requirements Identification Document

Each of the above listed documents were used to create the checklists cited for use in evaluating the implemented systems. These requirements were also used in evaluating the systems still being designed.

Table 13-1 lists the DOE orders and standards that apply to the design and operation of facilities. Through checklists, these orders and standards have been used to evaluate the current designs and to provide human factors design considerations into the design modification process.

13.3 HUMAN FACTORS PROCESS

The SNF Project recognizes the need for the inclusion of human factors principles in the design, operation, and maintenance of SNF Project facilities. The SNF Project is committed to including human factors principles through a process of expert review of the information requirements, job task knowledge, skills, and abilities, and control layouts, display relationships, gauging, instrumentation, physical access, and other HFE issues associated with the operation and maintenance of the SNF Project facilities.

The SNF Project's human factors process meets the requirements of DOE Order 6430 1A and implements it for SNF Project facilities. This process will ensure appropriate consideration of the HFE aspects involved with constructing, operating, and maintaining the facilities. This process utilizes existing procedures for ensuring that HFE issues are captured, tracked, and implemented with sufficient organizational control and adequate configuration management priority. The SNF Project human factors process is currently being updated and will adequately address the requirements of DOE Order 6430 1A.

HNF-PRO-229, Technical Procedure Standard, states that human factors are considered during procedure preparation to minimize risk to personnel and equipment, and procedures are written to the degree of detail necessary to safely perform the activity. HNF-PRO-704, Hazard and Accident Analysis Process, provides a process to systematically identify the hazards associated with a given operation and determine the engineered and administrative features taken to eliminate, control, or mitigate the identified hazards and evaluate their adequacy. Human–machine design considerations include communication, and operational aids, layout of controls and instrumentation and labeling, work environment factors such as heat, light, noise, physical access, protective clothing and breathing apparatus and demonstrated ability of personnel to accomplish their responsibilities under normal, abnormal, and accident conditions.
Table 13-1  U.S. Department of Energy Requirements and Human Factors Engineering Standards Matrix (2 sheets)

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<th>Applicable HFE implementation standards</th>
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<td><strong>Operational aids</strong></td>
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<td>DOE Order 5480 23 Attachment 1 14a</td>
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Note: NUREG-0800 Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. Standard Review Plan 181 requirements were reviewed against Section 1300 12 4 of DOE Order 6430 1A and Volume 1 (Draft) of DOE STD 1062 94 [Note: NUREG-0800 is the same document called out in HNF SD SNF DB-003 Item 9].


DOE Order 6430 1A General Design Criteria U S Department of Energy Washington D C.

DOE STD 1062 94 1994 Ergonomics and Human Factors Engineering Design Criteria Volume 1 General Criteria (Draft) U S Department of Energy Washington D C [Note: this is the same document called out in HNF SD SNF DB-003 Item 9].


DOE = U S Department of Energy

HFE = human factors engineering

NA = not applicable
SNF-4399, Spent Nuclear Fuel Project Human Engineering Program Plan (HEPP), addresses the human factors and ergonomics processes in detail. SNF-4399 meets the requirements of DOE Order 6430 1A, Section 1300, as applicable. The human factors process applied to the SNF Project facilities is described in the following sections.

13.3.1 Systematic Inquiry

DOE Order 5480 23 and DOE-STD-3009-94, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports, require a systematic inquiry into the HMI's pertinent to safety, especially focusing on those systems that are defined as safety SSCs. Those systems were evaluated using HFE checklists especially designed for HFE evaluations (INEL-95/0117). The checklists are designed to evaluate the HMI of safety-related systems according to criteria established by DOE, the U.S. Nuclear Regulatory Commission, and the HFE professional community. The checklists can be applied to both existing equipment and designs. The checklists result in one of four responses: (1) compliant with the criterion, (2) not compliant with the criterion, (3) the criterion is not applicable, and (4) to-be-determined as the design evolves. Once the HMI's are completed, the issues that resulted in to-be-determined responses are resolved so that the checklist items can classified as compliant, not compliant, or not applicable. For SNF Project facility design and construction, the noncompliance responses were grouped into either "to be resolved" or "not appropriate to the design being contemplated." The facility FSAR Annexes contain results of implementation inquiries. The INEL-95/0117 Human Factors Engineering Checklists for Application in the SAR Process, checklists are maintained in the SNF project files.

13.3.2 Lessons-Learned Applications

The HFE process begins by collecting "lessons learned" from experience in the commercial industry, DOE facilities, and relevant events in other technical domains. As discussed in Chapter 3.0, the dockets for Title 10, Code of Federal Regulations, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR 72), were reviewed, including notices of violations issued and reported under 10 CFR 72 75, "Reporting Requirements for Specific Events and Conditions." Events at DOE facilities similar to the SNF Project facilities also were reviewed. These reviews were conducted to ensure that any specific failure in the human–machine interaction that may be applicable to the design of SNF Project facilities were addressed. This also allowed the design teams to glean valuable lessons learned from the experience of others. SNF Project design engineers ensured that lessons learned applicable to maintenance and operation of the SNF Project facilities were discussed with the vendors.

Figure 1 of DOE-STD-1029-92 illustrates a facility safety envelope and shows how lessons learned contribute to human performance. Lessons learned flow into (i.e., are an input to) DOE safety policy, facility safety policy, bases for facility operations, and facility operations (technical safety requirements, training and qualifications and operations, maintenance, and surveillance).
procedures), which all contribute to human performance. Therefore, lessons learned are incorporated into all aspects of the design—from the regulatory requirements and design requirements to operating procedures and training requirements.

As discussed in Chapter 17, the SNF Project maintains a lessons-learned program that includes the identification, documentation, and dissemination of lessons learned. Each facility has a point of contact who is responsible for the distribution of lessons learned to the appropriate organizations or individuals within the facility. The sources of lessons-learned occurrences and conditions include U.S. Nuclear Regulatory Commission documents, Hanford Site internal lessons learned, Occupational Safety and Health Administration safety bulletins, operational readiness review final reports, and Operating Experience Weekly summaries. If applicable, procedures are modified, training is upgraded, and/or maintenance practices are revised to incorporate lessons learned and thereby avoid a recurrence of an adverse work practice or operating experience. The overall lessons-learned process for the SNF Project is described in SNF Project administrative procedures.

13.3.3 Engineering Design and Analysis

Engineering design encompasses design process and analysis in support of original design and design change activities. Designs are prepared in accordance with criteria established in DOE Order 6430.1A, Section 1300-12, in which human factors are considered through the system development process (planning, requirements analysis, system design, and system testing and evaluation). Pursuant to DOE Order 6430.1A, Section 1300-12.4, the design, operation, and maintenance of SNF Project facilities considers appropriate human factors technology to improve human performance through enhancements in the work environment and HMIs. HFE design is applied at a level commensurate with the safety importance, complexity, and degree of HMI of the SSC.

The design and design specifications for SSCs were reviewed according to the DOE orders listed in Section 13.2.

HFE aspects have been included within the design requirements at each step. A separate HFE program has not been developed because it has been determined that, for the purposes of developing requirements and ensuring their appropriate inclusions, the project engineers responsible for the design should have responsibility for the applicable HFE requirements. This process of placing HFE responsibility on the design engineers is similar to promoting accountability downward and reaping the benefits of concurrent HFE in the design phases. In addition, all human factors requirements were made part of the SNF Project procurement for SSCs. Using this approach, SNF Project personnel are aware of applicable HFE criteria, and their design reports and activities make reference to HFE considerations. See Figure 13-1 for a diagram of the process for incorporation of HFE concepts into the SNF Project designs. See the facility FSAR Annexes for results of facility-specific HFE analyses.
Testing and evaluation is considered during verification and validation of systems. Human factors considerations were addressed during testing. See the facility FSAR Annexes for specific human factors interactions.

13.3.4 Designing for Abnormal Operations

An assessment team conducted a hazard analysis at each SNF Project facility to identify potential hazards to the facility worker, collocated workers, the public, and the environment. Potential events considered ranged from normal operation through beyond design basis accidents. The design basis accidents were considered to the extent described in Chapters 3.0 of the facility FSAR Annexes. The hazard analysis process includes identification and analysis of the various operational steps to be conducted within the SNF Project facilities. Human interactions with machines are considered in the analyses. The hazard analysis identifies both engineered and administrative safety features. Engineered safety features are design features that prevent or mitigate a potential accident and may require HMIs. Administrative safety features are procedures and programs designed to ensure that engineered safety features continue to be available for accident prevention or mitigation. Administrative safety features also include programs that implement a particular set of safety practices. The hazard analysis methodology and results are described in more detail in Chapters 3.0 of the facility FSAR Annexes.

The HMIs for engineering safety features were analyzed according to the criteria established in Section 13.2. The analysis results are summarized in the facility FSAR Annexes.

13.4 IDENTIFICATION OF HUMAN–MACHINE INTERFACES

Summaries of safety-class and safety-significant SSCs and the associated HMIs are provided in the facility FSAR Annexes.

13.5 OPTIMIZATION OF HUMAN–MACHINE INTERFACES

Discussions of the optimization of HMIs are provided in the facility FSAR Annexes. The following subsections discuss how the various aspects of HFE are implemented in the SNF Project to assure successful operations.

13.5.1 Human Factors Engineering Analysis of Human-Machine Interfaces

In addition to the HFE review of SNF Project facility systems and the general physical and organizational environment, normal operations were reviewed for the potential for human error and for the programmatic issues related to training, staffing, technical procedure development, and maintenance activities. This review process is depicted in Figure 13-2.
Tabletop task analyses, systems reviews, direct interviews with design authorities and cognizant engineers, and reviews of applicable design documentation were conducted to evaluate whether the design and procurement meet the established HFE guidelines and standards. The results of the task analyses, interviews, and design documentation reviews were analyzed using INEL-95/0117. These checklists are used to determine whether designs meet applicable HFE requirements, whether the design specifications account for the HFE requirements, or whether the HFE requirement is to be considered in the design modification process. Finally, the facility hazard analyses were reviewed to determine whether human operator actions involve using equipment or taking action that must be further analyzed. The checklists (designed per INEL-95/0117) are maintained in the SNF project files.

13.5.2 Staffing

13.5.2.1 Operator Capabilities  Skills and standards that must be attained for a particular job are defined to ensure that personnel have the appropriate knowledge, skills, and abilities to perform the required activities in a safe and reliable manner. As stated in Chapter 12.0, minimum education and experience requirements for management and technical personnel meet the requirements of the SNF Project facility standards/requirements identification document (S/RID) (HNF-SD-SNF-RD-001). Qualification criteria are established to define the experience, education, and training required to perform a designated job. Chapter 5.0 also provides a discussion of the minimum staffing requirements.

A facility-specific training program is developed and maintained as a performance-based training program. Several positions require certification in accordance with the SNF Project facility S/RID (HNF-SD-SNF-RD-001) as fissionable material handlers. These positions are identified in the project training implementation matrix.

Operator capabilities are verified throughout testing and review programs and walkthroughs. Specifically, the operational readiness reviews will include dry run demonstrations that operators and procedures are in a satisfactory state of readiness to safely and efficiently perform the required operations. The dry run demonstrations validate both procedures and the operations staff by confirming that personnel have the appropriate knowledge and capabilities to perform their tasks. Therefore, the SNF Project training plan is considered adequate with respect to the inclusion of human factors concerns.

The FSAR process for selecting safety-class and safety-significant SSCs and technical safety requirements also serves as a systematic inquiry into the ability of facility staff to accomplish their responsibilities and duties during normal and abnormal operations. Input is provided from various disciplines (including operations) in the development of the hazard analysis. For each analyzed accident scenario, operators and engineers are involved in the FSAR process for selecting safety-class and safety-significant SSCs and technical safety requirements for facility operation.
13 5 2 2 Staffing Plan  A current staffing plan through 2002 exists for the SNF Project. A review of this plan concluded that appropriate consideration has been given to the following:

- Learning curves
- Loss of trainees through the process
- Appropriate job titles and functions for operations personnel

Facility-specific staffing levels are addressed in the facility FSAR Annexes.

13 5 3 Procedures

The SNF Project is committed to the inclusion of human factors principles in the development of technical procedures. SNF Project personnel have developed standards and attended training to ensure that technical procedures will follow human factors principles in their content, layout, and format.

The SNF Project has complied with DOE Order 5480 19, DOE-STD-1029-92, HNF-PRO-229, and Fluor Daniel Hanford, Incorporated, contractual obligations. Acceptable compliance is demonstrated by a writer’s guide meeting at least the requirements in DOE Order 5480 19 and DOE-STD-1029-92. A review of TP-DI-001, SNF Project Technical Procedures Writer’s Guide demonstrates that the SNF Project meets the technical procedure writing requirements of DOE Order 5480 19 and DOE-STD-1029-92. TP-DI-001 was developed at the Hanford Site for technical procedure writers. TP-DI-001 includes the technical procedure writing requirements of DOE-STD-1029-92. The Human Factors Area Council created DOE-STD-1029-92 to ensure writers of technical documents will follow basic human factors principles in their content, layout, and format of written technical procedures. DOE-STD-1029-92 complies with accepted human factors principles. Inclusion of the Standard demonstrates that human factors considerations are included in the technical writers’ work.

DOE-STD-1029-92 is used in its entirety to provide guidance to technical procedure writers.

SNF Project technical procedure writers have attended training provided by several offsite experts in technical procedure writing. Subject matters covered by these experts included grammar, layout, format, and human error reduction. Training records exist for SNF Project procedure writer’s completion of these courses. A training matrix that specifies when retraining is necessary is maintained for each procedure writer.

Each technical procedure being developed for startup and operation of all SNF Project facilities is being written to the TP-DI-001 and by writers that have been appropriately trained in the inclusion of human factors considerations, as described in DOE-STD-1029-92. Procedures play a key role in ensuring safe, deliberate, and controlled operations. The intent of DOE-STD-1029-92 is to provide a knowledgeable and reasonable approach to all aspects of procedure management that will ensure the availability and use of sound procedures at DOE facilities.

DOE-STD-1029-92 provides a method for writers to ensure key questions are
addressed and that procedures contribute to maintaining safe operations. For example, key human factors questions that procedure writers are trained to answer include:

- Who is the user and what is the user’s level of experience and training?
- How does this document relate to other procedures for the equipment and facility?
- What materials, equipment, and facilities are to be used?
- What tasks are to be accomplished?
- Why, when, where, and how are the tasks to be accomplished?

Procedure writers learn that the operators are an important source of information when developing procedures. Personnel from the operating organization should be involved in the process from the initial decision to write a procedure through the review, verification, and validation of the procedure.

Facility operators can provide valuable insight into the job being performed, information about past operating experience, and data for developing the basis of the procedure. Their involvement in the process can include participating in walkthroughs and identifying behavioral obstacles, cautions, and valuable nonstandard source documentation beyond safety analysis reports, probabilistic risk assessments, technical safety requirements, vendor manuals, and old procedures.

The procedure has to be constructed and written to communicate to the procedure user all information needed to successfully operate the system within its design environment. Human factors considerations would include such things as using simple and concise language, using action statements, maintaining consistency in language and format among instrument labeling, procedures, and training, and avoiding ambiguity. Limiting use of abbreviations and acronyms, and adhering to grammatical conventions and syntax of standard American English.

The procedure is verified and validated during the test and evaluation of the system. Therefore, the inclusion of human factors principles is occurring and will continue to occur throughout the technical procedure writing process. These actions are considered adequate for compliance to human factors requirements.

In addition to training or day-to-day supervision, providing sound procedures and requiring workers to use them are among the most formal, direct, and effective methods available to facility managers to ensure that operations meet DOE’s objectives. Procedures provide managers with a critical tool to communicate detailed expectations for how individual workers are to perform specific tasks. In other words, the facility’s operating procedures provide the mechanisms for ensuring that facility operation is maintained within the safety basis.

All facility activities are governed by established procedures that specify specific sequences of steps that must be performed to accomplish a desired task. As described in Chapter 12.0, the technical basis for procedures is determined by performing a task analysis on each step in the process. Furthermore, the procedure system requires end-user (operator) participation in both the procedure writing and the review processes. This participation permits the inclusion of...
operational experience and human factors considerations into procedures and ensures that human needs and limitations are carefully considered and incorporated.

Chapter 12 describes the validation portion of the review process for draft procedures, during which the procedure is tested for usability, correctness, compatibility with the system equipment, and human factors considerations incorporation. Validation includes determining whether the user can perform each step correctly and ensuring the units used in the procedure are consistent with the facility instrumentation and possess the required precision for operation. Validation ensures that the sequence of steps is logical and can actually be performed, all documentation (e.g., labels, equipment representations, and diagrams) contained in the procedure is accurate, and tasks required by the procedure are within the operator's capabilities. Each step in each procedure will be checked against the actual equipment and controls to ensure the step is technically accurate. Human factors concerns will also be incorporated from feedback from the operators about the procedures and by observing the operators' interactions with the actual equipment. The goal is to ensure that the procedure can be effectively used by operations personnel during normal, abnormal, and emergency conditions. Alarm response procedures are developed on the basis of the monitoring and control functions of the system or component providing the alarm output. All operating procedures are required to be validated prior to use.

As discussed in Chapter 12, procedures will be periodically reviewed. This periodic review process ensures the technical accuracy and the proper consideration of human factors issues in procedures. Operating procedures, alarm response procedures, and surveillance and test procedures are included in the review process. The process involves verification of the technical basis and procedural steps, validation of the usability of the procedure by operations personnel, and verification of human factors elements that may impact the effectiveness of the procedure.

SNF-4399 addresses the human factors and ergonomics process in detail, including HFE and ergonomics testing and validation. SNF-4399 meets the requirements of DOE Order 6430 1A, Section 1300, as applicable.

13.5.4 Training

Training ensures that all staff have the knowledge, skills, and abilities required to safely operate and maintain the facilities and processes. As discussed in Chapter 12, the SNF Project training organization applies performance-based training to specific job requirements. This training will be implemented using a systematic approach to training development that is in compliance with the requirements of the SNF Project facility S/RID, Chapter 4 (HNF-SD-SNF-RD-001).

Operators receive on-the-job training that consists of learning to operate the specific components and systems. It is anticipated that the majority of the on-the-job training package development and operator evaluations will be performed during preoperational testing. As previously stated, preoperational testing includes dry runs using mockups and in situ equipment and systems to provide verification of the operators' knowledge of facility procedures.
readiness of tools, equipment, and instrumentation, and off-normal and emergency operations.

The major focus is the incorporation of job and task analysis information into the on-the-job training material and ensuring that operators are trained and evaluated on this material.

For some tasks, the job analysis or task analysis will identify the need for continuing training. Schedules, materials, and evaluations for continuing training will be developed. This may include (1) training on selected fundamentals, systems, administrative controls and emergency response and (2) on-the-job training for normal and abnormal operation. Job and task analysis also assists in defining those tasks that require continuing training. To ensure that training properly reflects the current operating practices, conditions, and procedures, a training materials maintenance system is used to track items that may affect the content of training programs. Chapter 120 provides additional details on the maintenance of training programs and the use of a training materials maintenance system.

Operational drills and exercises are an integral part of the facility training program. Drills are an effective means of providing training in the handling of abnormal conditions and situations. Drills also provide management with an assessment tool to help evaluate the staff's ability to respond to off-normal events and conditions. Drills and exercises provide the means for testing crew proficiency and validating procedures. At the completion of operational testing, personnel will have the appropriate training that includes human factors considerations.

13 5 5 Maintenance

Vendor contracts include development of appropriate maintenance manuals and listings of common maintenance to be performed on the systems. Normal maintenance, including preventive maintenance, will be accomplished by SNF Project staff. Operations planning accounted for personnel and equipment necessary to perform maintenance activities. Maintenance activities include ensuring that appropriate diagnostic equipment and maintenance tools are available and the staff has adequate skills and training to accomplish the necessary activities. The above actions are considered acceptable by the human factors professional.

13 5 6 Abnormal Operations

Scenarios for abnormal operations involving human recovery initiator or mitigator actions are addressed in Chapters 30 of the facility FSAR Annexes. Recovery actions that require human intervention initiator or mitigator responses involve the equipment analyzed in Sections 13 5 of the facility FSAR Annexes. Human factors considerations for abnormal operations are covered in Sections 13 5 3 of the facility FSAR Annexes.
13.6 REFERENCES


DOE Order 6430 1A, *General Design Criteria*, U S Department of Energy, Washington, D C


HNF-PRO-704 *Hazard and Accident Analysis Process*, Fluor Daniel Hanford, Incorporated, Richland, Washington


fsar13 sar 13-14 November 1999


TP-DI-001, *SNF Project Technical Procedures Writer s Guide*

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DRD = design requirements document
HFE = human factors engineering
RFP = request for proposal
S/RID = standards/requirements identification document
SSC = structure, system, and component
Figure 13-2 Human Factors Engineering Methodology for Design Review

Familiarization with SSCs

Review of the facility hazard evaluation

Selection of human machine interfaces to be evaluated based upon those systems that fall into one of two categories: (1) safety-class SSCs or (2) pertinent to the effective operation.

Tabletop task analysis to determine the tasks to be performed the sequence of operations and maintenance activities.

Interviews with cognizant project personnel to determine the current level of the design and anticipated design efforts.

Review the design requirements process pertinent subcontract RFPs and overall facility design process for inclusion of human factors requirements.

Application of HFE checklists to existing SSC and design requirements documents.

Checklist evaluation results sorted into one of four categories: (1) compiled, (2) not compiled, (3) not applicable, and (4) TBD.

Resolve any noncompliances.

Gather commitments to consider TBD items in further design efforts.

Review programmatic issues (staffing, maintenance procedures, and training).

Resolve any programmatic concerns.

Report on the results of the HFE evaluation process.

HFE = human factors engineering
RFP = Request for Proposal
SSC = structure system and component
TBD = to be determined
CHAPTER 140

QUALITY ASSURANCE
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14 0 QUALITY ASSURANCE

14 1 INTRODUCTION

The Spent Nuclear Fuel (SNF) Project quality assurance program provides assurance that the design, procurement, construction, SNF characterization and conditioning, testing, inspection, operation, maintenance, packaging, handling, transportation, and interim storage activities conducted at the SNF Project facilities conform to regulatory and contractual requirements. This chapter summarizes the SNF Project quality assurance program for activities conducted at the SNF Project Facilities. It also provides references to documents containing additional details of the SNF Project quality program.

The SNF Project shall execute the applicable quality program requirements in a graded approach to an extent that is commensurate with the importance to safety of the item or related activity. Specific safety structures, systems, and components (SSCs) along with related activity classifications are presented in either the safety equipment list or the SNF Project 'Q' list (HNF-SD-SNF-RPT-007) as described in HNF-SD-SNF-RD-001, Spent Nuclear Fuel Project Standards/Requirements Identification Document. The SNF Project SSCs and related activities are described by three different safety classifications: safety class, safety significant, and general service. Quality-affecting activities performed by any organization that provides equipment, services, or support to the SNF Project facilities also are described in this chapter.

Adherence to the quality assurance program ensures that:

- U.S. Department of Energy (DOE) missions and objectives are effectively accomplished.
- Products and services are safe, reliable, and meet or exceed the requirements and expectations of the user.
- Any hazards to the public, to Hanford Site and facility workers, and to the environment are minimized.

14 2 REQUIREMENTS

Quality assurance requirements based on Title 10, Code of Federal Regulations, Part 830 120, "Quality Assurance Requirements" (10 CFR 830 120), and DOE O 414 1, Quality Assurance, define the quality implementing requirements for the SNF Project SSCs and activities. In addition, DOE-directed compliance with DOE/RW-0333P, Quality Assurance Requirements and Description (QARD), quality program requirements are applicable, as appropriate to certain SNF Project items and activities.
The Office of Civilian Radioactive Waste Management requirements in DOE/RW-0333P, as a minimum, apply to the following SNF Project activities, as they relate to disposal in the repository:

- Characterization or data collection for input or use
- Conditioning into final form
- Handling, packaging, and transportation

Requirements invoked by the standards/requirements identification document (S/RID) comprise the total list of environment, safety, health, and quality requirements to be implemented by a site, facility, or activity. These requirements are appropriate to the life cycle phase to achieve an adequate level of protection for worker and public health and safety and the environment during design, construction, operation, decontamination and decommissioning, and environmental restoration. S/RIDs encompass quality, health, safety, environmental and 'safety-related' safeguards, and security standards/requirements.

The Fluor Daniel Hanford, Incorporated contract S/RID contains standards and requirements, applicable to Fluor Daniel Hanford and its subcontractors necessary for safe operation of Project Hanford Management Contract facilities. Facility S/RIDs contain standards/requirements applicable to a specific facility that are the direct responsibility of the facility manager. Specific S/RID requirements are presented in HNF-SD-SNF-RD-001, *Spent Nuclear Fuel Project Standards/Requirements Identification Document*.

DOE has established a regulatory policy (Grumbly 1995) that SNF Project facilities achieve nuclear safety equivalency to comparable U.S. Nuclear Regulatory Commission (NRC)-licensed facilities. This is to be applied to matters of nuclear safety (including radiological control issues) for new SNF Project facilities. The NRC's guidance for the use of 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," is presented in NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities*, which states that SSCs are to be placed into two categories according to their function. They are classified as important to safety, or not important to safety. NUREG-1567 references NUREG/CR-6407, *Classification of Transportation Packaging and Dry Spent Fuel Storage Systems Components in Accordance with Importance to Safety*. NUREG/CR-6407 contains a graded approach for determining the classification of SSCs according to important to safety considerations:

- **Category A - Critical to Safe Operation**: SSCs in this category include those whose failure or malfunction could directly result in a condition adverse to public health and safety.
- **Category B - Major Impact on Safety**: SSCs in this category include those whose failure or malfunction could directly result in a condition adversely affecting public health and safety. The failure of a Category B item in conjunction with the failure of an additional item, could result in an unsafe condition.
• Category C - Minor Impact on Safety  SSCs whose failure or malfunction would not significantly reduce the packaging effectiveness and would not be likely to create a situation adversely affecting public health and safety.


• Safety Class - SSCs needed for public protection

• Safety Significant - SSCs of particular importance to defense-in-depth or worker safety

• General service or non-safety class - Little or no impact on worker or public safety

The quality assurance program requirements to be implemented for new SNF Project facilities will be determined by using a merged classification system of the DOE and NRC safety classification systems. The merged quality assurance requirements are as follows:

• For Category A important-to-safety SSCs, the DOE quality assurance program used for safety class SSCs will be implemented using 10 CFR 830.120, "Quality Assurance Requirements," HNF-MP-599, Project Hanford Quality Assurance Program Description, and DOE Order 6430.1A, General Design Requirements. Implementation of NRC equivalence (10 CFR 72, Appendix F, "General Design Requirements") is achieved in accordance with WHC-SD-SNF-DB-002 and HNF-SD-SNF-DB-003.

• For Category B important-to-safety SSCs, the DOE quality assurance program used for safety-significant SSCs will be implemented using a graded approach, in accordance with 10 CFR 830.120 and HNF-MP-599, which uses national and commercial procurement specifications for the implementation of the design. The SNF Project implements the safety-significant classification as defined in HNF-PRO-704, Hazard and Accident Analysis Process.

• For Category C important-to-safety SSCs, the DOE quality assurance program used for general service SSCs will be implemented using good engineering practices and national and commercial standards to control design and procurement.

An evaluation, documented in WHC-SD-SNF-DB-002, Spent Nuclear Fuel Project Path Forward, Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities, identifies requirements to establish nuclear safety equivalency that are to be met in addition to existing and applicable DOE requirements. These requirements, except those related to the design basis earthquake, are contained in HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward.

The integrated quality assurance requirements for the SNF Project are summarized in Figure 14-1, which depicts the SNF Project quality assurance program document hierarchy.

The specific regulatory requirements applicable to this chapter include:

- **DOE O 414 1, Quality Assurance**
- **DOE/RW-0333P, Quality Assurance Requirements and Description**

Applicable facility design codes, standards, statutes, rules, regulations, and DOE Orders are summarized in the following technical documents:

- **HNF-MP-003, Integrated Environment Safety and Health Management System Plan**
- **HNF-SD-GN-TI-502, Project Hanford Nuclear Facilities List and Authorization Basis Information**
- **HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward Additional NRC Requirements**
- **HNF-SD-SNF-DB-005, Spent Nuclear Fuel Project Multi-Canister Overpack Additional NRC Requirements**
- **HNF-SD-SNF-RPT-007, Application of the Office of Civilian Radioactive Waste Management Quality Assurance Requirements to the Hanford Spent Nuclear Fuel Project**
- **WHC-SD-SNF-DB-002, Spent Nuclear Fuel Project Path Forward Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities**
Additional requirements applicable to specific SNF Project facilities are presented in the facility annexes to the SNF Project Final Safety Analysis Report (FSAR). These documents provide the quality assurance requirements for federal repository acceptance of SNF and satisfy U S Nuclear Regulatory Commission equivalency requirements.

Quality assurance program plans (QAPPs) identify or reference the management processes, functional responsibilities, interfaces, and procedures for implementing HNF-MP-599, *Project Hanford Quality Assurance Program Description*, and project-specific quality assurance requirements.

14.3 QUALITY ASSURANCE PROGRAM ORGANIZATION

14.3.1 Program

The Project Hanford quality assurance program is composed of the Fluor Daniel Hanford quality assurance policy, HNF-MP-599, the quality assurance program implementation plan for nuclear facilities, Project Hanford SNF QAPPs, subcontractor QAPPs, and Fluor Daniel Hanford and subcontractor implementing procedures and work control documents. Figure 14-1 depicts the Project Hanford quality assurance program document hierarchy. The QAPPs, as shown in Figure 14-1, describe the respective quality assurance programs, which present criteria, requirements, responsibilities, interface relationships, and implementation documents for the SNF Project, including facilities and activities.

The QAPPs are maintained to ensure compliance with requirements specified in 10 CFR 830 120, DOE O 414 1, DOE/RW-0333P, 10 CFR 71, Subpart H, and 10 CFR 72, Subpart G, as implemented in HNF-MP-599.

The SNF Project organization that will be implementing the quality assurance program is described in Chapter 17.0, which includes a description of the organizational structures, responsibilities, authorities, and interfaces that meet the requirements of Section 1 of DOE/RW-0333P 10 CFR 72 142, and 10 CFR 830 120.

The SNF Project QAPP quality assurance policy statement includes the following:

- Compliance with the provisions of the QAPP is mandatory.
The SNF Project has committed to implement an effective quality assurance program that is in compliance with the applicable DOE and federal regulatory requirements as presented in Section 14.2.

- Quality assurance controls are written into SNF Project implementing procedures that are used to control work processes.
- Personnel shall be appropriately trained to the procedures they will use prior to the start of the work.
- No work subject to the requirements of the governing documents shall be started prior to the development, review, approval, and issuance of the appropriate work documents or procedures.
- Achievement of quality is a line responsibility, where each performer is accountable for the quality of the work assigned. Line organizations and line management are responsible for the effectiveness of the quality program for SNF Project facilities.

The QAPP establishes prime responsibility for the implementation of the quality program to the SNF Project management and the managers of those organizations that support the SNF Project facilities.

The degree of control over activities affecting the quality of items and services is commensurate with the importance to safety and, as necessary, to ensure conformance to the approved SNF Project facility design. SNF Project personnel have the authority to stop work in accordance with approved procedures for unsafe situations and to control further operation until the conditions that created the unsafe conditions are corrected. Stop work actions are immediately communicated to project management.

Suppliers and subcontractors providing services for the SNF Project are required to implement a quality assurance program consistent with the requirements specified in the SNF Project procurement documents. Quality assurance programs used by subcontractors providing safety-related supplies and services are submitted to the SNF Project for review and approval.

Operational readiness reviews are performed in accordance with approved procedures for determining compliance of the SNF Project facilities and activities with applicable requirements prior to startup or restart of the facility or process.

Compliance with the QAPP is achieved by implementing written procedures that are identified in the QAPP within a baseline requirements matrix, supplemental baseline implementation matrix, and/or quality assurance program index. The QAPP presents the overall structure of the documents hierarchy to meet the quality program requirements.
Application of project activity procedures is described in the QAPP. Procedures identified in a baseline implementation matrix are applied to all SNF Project elements unless exceptions are justified. A supplemental baseline implementation matrix identifies additional procedures to implement SNF subproject-specific requirements. Requirements have been identified that shall be applied to SSCs in specific SNF Project facilities that must also comply with DOE/RW-0333P, 10 CFR 71, Subpart H, and/or 10 CFR 72, Subpart G. Implementing procedures for these requirements are presented in a quality assurance program index. These procedures are used in addition to those identified in a baseline implementation matrix and the applicable procedures identified in a supplemental baseline requirements matrix for the particular SNF Project facility.

The SNF Project will ensure that the quality assurance program requirements will remain in effect during the lifetime of the SNF Project facilities through maintenance of the QAPP and implementing procedures.

14.3.2 Personnel Training and Qualification

Personnel performing activities affecting quality are trained and qualified to perform assigned tasks. This includes knowledge of the work processes, tools, equipment, and requirements. The training and qualification program provides for the development of personnel proficiency commensurate with the scope, complexity, and nature of an assigned activity. Training and indoctrination needs are identified and documented by management.

SNF Project management and managers of those organizations that support the SNF Project facilities are responsible for ensuring that their workers are sufficiently trained to perform assigned tasks in a manner that minimizes (1) risk to the worker performing the task, coworkers, and the public, (2) negative impacts to the environment, and (3) risk of damage to the facility and facility equipment. Training and qualifications for specified job functions are based on analysis of the specified duties and tasks associated with the functions. Job functional descriptions are developed that describe minimum requirements for education, experience, and when required, physical condition, and certification. Management is responsible to verify qualification of personnel before assigning them to do work. Inspection and nondestructive examination positions are certified in accordance with written procedures that are based on ASME NQA-1-1994, Quality Assurance Program Requirements for Nuclear Facilities, or other standards applicable to the work performed.

Qualification and training requirements are periodically reviewed to ensure that they continue to reflect the current systems, procedures, and policies applicable to each position. Continuing training is provided to ensure that job proficiency is maintained. Training is provided by instructors having the technical and instructional skills necessary to provide the training in an effective manner. Training program effectiveness and efficiency is determined by training management through feedback from instructors, students, students' managers, and periodic reviews. Improvements are addressed to increase proficiency based on existing or new requirements.
The SNF Project personnel training and qualifications process meets the requirements of Part 2, Section 2, "Personnel Training and Qualifications," of HNF-MP-599. Implementing procedures for personnel selection, indoctrination, training, and qualification are identified in the respective matrix associated with each SNF Project QAPP. Specific training and qualification requirements for personnel who perform or manage Office of Civilian Radioactive Waste Management-related design, scientific investigation, software development activities, and for personnel who verify or manage the verification of design, scientific investigation, software development activities or items, shall meet the requirements specified in DOE/RW-0333P, Section 2.2.12.

### 14.4 QUALITY IMPROVEMENT

The objective of quality improvement is to detect and correct problems adversely impacting quality and to continuously improve the quality of items and work processes. The basis of the approach to quality improvement is that (1) work activities can be planned, performed, assessed, and improved, and (2) lessons learned from this process can be used when planning subsequent activities to preclude recurrence of problems.

Under the quality improvement program, organizations have the authority to identify quality problems and to initiate, recommend, or provide solutions through designated channels. Conditions adverse to quality (e.g., failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances) are promptly identified to the appropriate level(s) of management and corrected. Management systems for quality improvement include problem identification, screening, evaluation, reporting and resolution, nonconformance identification, control and tracking, corrective action process including root cause analysis, trending, and tracking problems to closure.

Item characteristics, process implementation, and other quality-related information are reviewed and data analyzed to identify items, services, and processes needing improvement. This review and analysis includes an assessment of risk. Items, services, and processes that do not meet established quality requirements are identified, controlled, and corrected promptly in accordance with the significance of the problem and work. Verification of corrective action completion is also conducted.

A problem identification process is used to identify problems that are then evaluated for an unreviewed safety question and Price-Anderson Amendments Act of 1988 violation subject to Title 10, Code of Federal Regulations, Part 820, "Procedural Rules for DOE Nuclear Activities" (10 CFR 820). Correction includes identifying the cause of the problem, risk assessment, determining corrective action for the problem, problem closure, and initiating any additional corrective actions to preclude recurrence.

Periodic inspection of SSCs and equipment is performed to determine whether deterioration is taking place and to identify and address obsolescence that threatens personnel safety or facility preservation.
Performance indicators, item characteristics, process implementation, corrective actions assessments, and other quality-related information are reviewed and data analyzed to identify items, services, and processes needing improvement. When appropriate, these improvement evaluations include use of tools such as root cause analysis and lessons learned. Problem identification processes and nonconformances are periodically analyzed to identify quality trends.

The SNF Project quality improvement process meets the requirements of Part 2, Section 3, "Quality Improvement," of HNF-MP-599. Implementing procedures for quality improvement requirements are identified in the respective matrix associated with each SNF Project QAPP.

14.5 DOCUMENTS AND RECORDS

The documents and records program establishes requirements for control of the preparation, review, approval, issuance, use, and revision of documents that prescribe processes, specify requirements, or establish design. Controlled documents are reviewed for adequacy, completeness, and correctness prior to approval for use. Controlled document revisions are reviewed for adequacy, completeness, and correctness before approval by the organization that originally reviewed and approved the document, or by a designated alternate organization with comparable technical competence and capability. After approval, controlled documents are released to specified users to ensure the latest approved revisions are available to personnel for use at the location where the work is being performed. Compliance to these controlled documents is required. Document control is also discussed in Chapter 17.

Records are specified, prepared, reviewed, approved, maintained, and stored in accordance with approved procedures and instructions. Records retention requirements are specified in the procedures that govern generation of the records.

Records are to be legible, identifiable, accurate, complete, protected, and retrievable. Maintenance of records includes provisions for correction, replacement, retention, preservation, traceability, and accountability. Computer hardware and software used to maintain, index, store, or access records are maintained and controlled to ensure accountability, reproducibility, and protection from loss.

The SNF Project documents and records process meets the requirements of Part 2, Section 4, "Documents and Records," of HNF-MP-599. Implementing procedures for document and record requirements are identified in the respective matrix associated with each SNF Project QAPP.

14.6 QUALITY ASSURANCE PERFORMANCE

This section provides an overview of the SNF Project process to ensure that the performed work meets requirements. Work processes, design, procurement, installation and testing processes, handling, packaging, operation, maintenance, and interim storage are covered in Sections 14.6.1 through 14.6.4.
14.6.1 Work Processes

Work processes include, but are not limited to, activities involving design, analysis, fabrication, procurement, construction, handling, shipping, storing, cleaning, assembly, inspection, installation, and testing. Work on the SNF Project facilities is planned, authorized, and performed under controlled conditions in accordance with approved technical standards and administrative controls using approved procedures, instructions, plans, or other control documentation commensurate with the complexity and risk posed by the work to be performed. Such procedures, instructions, plans, or other control documents contain or reference all of the necessary administrative and technical requirements, including the sequence of actions and interactions, as required to ensure that activities are properly performed and meet acceptance criteria or other requirements.

Work process documents are developed, reviewed, and approved by personnel technically knowledgeable of the work. Work process documents are readily accessible to the worker. Supporting documentation for work activities is reviewed to ensure that the desired quality is being maintained and to identify areas for improvement. Use of such controls is to ensure that process parameters are controlled within defined limits and that specified environmental conditions are maintained. Work completion is documented and appropriate records maintained. Special processes (e.g., welding, heat treating, nondestructive testing) are performed by qualified subcontractors. Specific requirements are established in procurement documents.

Identification is maintained on items, or in documents traceable to them, in a manner that ensures identification is established and maintained to ensure control and maintenance of items for manufacture or receipt through delivery, installation, or use. Items may include hardware, samples, or data. This item identification and administrative control is used to prevent the use of incorrect or defective items and to maintain traceability for items, as required by specifications, codes, and standards. Handling, marking, storage, packaging, shipping, cleaning, and preservation of materials and other items is controlled to prevent damage, loss, or deterioration. Marking and labeling of items is maintained throughout the processes of packaging, shipping, handling, and storage. Status indicators will not be detrimental to the item.

Calibration and maintenance of equipment used for process monitoring and data collection are conducted in accordance with approved procedures. Computer software used in applications important to environmental, safety, health, and quality aspects shall be subject to appropriate controls throughout the software's life cycle.

The SNF Project work process meets the requirements of Part 2, Section 5, "Work Processes" of HNF-MP-599. Implementing procedures for the requirements are identified in the respective matrix associated with each SNF Project QAPP.

14.6.2 Design

Sound engineering and scientific principles, codes, standards, and practices for the assurance of technical quality are identified and incorporated into the design of new or
The design, including design changes, incorporates applicable design requirements, including design bases, functional and performance requirements, regulatory requirements, codes, standards, environmental conditions, and interfaces with new or existing items. DOE has established a regulatory policy (Grumbly 1995) that new SNF Project facilities achieve nuclear safety equivalency with U.S. Nuclear Regulatory Commission-licensed facilities.

The SNF Project design process meets the requirements of Part 2 Section 6 "Design," of HNF-MP-599 Implementing procedures for design are presented in the QAPP.

### 14.6.2.1 Design Inputs

Design, including design changes, incorporates design inputs consisting of controlled requirements, customer expectations, design assumptions, applicable design requirements, including design bases, functional and performance requirements, federal codes, DOE orders and standards, conceptual design criteria, national standards, specifications, drawings, environmental conditions, health and safety considerations, expected life cycle, reliability requirements, and interfaces with new or existing items. Design inputs are identified and documented and their selection reviewed and approved by the responsible design organization to ensure that sound engineering and/or scientific principles and appropriate standards are being used.

A design authority will be assigned for each SSC that will establish and document the design baseline(s) for each assigned SSC. The chief engineer will formally approve the assignment of each design authority. A design authority is responsible for:

- Maintaining the design baseline consistent with the physical configuration of the SSC(s) to ensure that the design baseline is technically correct and meets design requirements.
- Approving modifications to an existing design baseline, establishing an existing design baseline, or establishing a new baseline.
- Establishing design requirements and ensuring that design documents accurately reflect the design baseline.
- Informing affected organizations (e.g., Safety, Quality Assurance, Operations) of changes.

The engineering change notice process is described in Section 14.6.2.4 Computer software validation requirements are contained in Section 14.6.2.6.

### 14.6.2.2 Design Process

Appropriate quality standards are identified and documented as design input and their selection reviewed and approved. Design methods, materials, parts, equipment, and processes that are essential to the functions of the items affecting quality are selected and independently reviewed for suitability of application. Identified deviations from quality standards are controlled in accordance with procedures or instructions.
The final design of SNF Project facilities is related to the design bases input by documentation of sufficient detail to permit design verification, and by identification of assemblies or components that are part of the item being designed. Design documents are adequate to support facility design evolution, procurement, construction operation, and deactivation. The documentation references include the applicable codes, standards, and practices in addition to the applicable regulatory safety requirements.

Design interfaces are identified and controlled and design efforts coordinated among the participating organizations. Interface controls are documented and include the assignment of responsibility and the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces. Design information transmitted across organizational interfaces is documented and controlled. Transmittals identify the status of the design information or document provided and, where necessary, identify incomplete items that require further evaluation, review, or approval.

Use of commercial-grade items will be accomplished in accordance with documented processes using recognized nuclear industry standards.

14.6.2.3 Design Verification and Validation Design verification or validation of the adequacy of design products is accomplished by technically knowledgeable persons other than those who performed the work.

Design adequacy is verified in accordance with approved procedures by individual or interdisciplinary design reviews, alternate calculations to verify the correctness of the original design calculations, or qualification testing to demonstrate adequacy of performance under conditions that simulate the most adverse design conditions. The extent of verification is based on the complexity, risk, and uniqueness of the design.

Design verification is performed in a timely manner and identified design errors corrected prior to approval and implementation of the design. Design verification for the level of design activity accomplished is performed prior to release for procurement, manufacture, construction, or release to another organization for use in other design activities except in those cases where this timing cannot be met, such as when insufficient data exist. In cases where insufficient data exist, the unverified portion of the design is identified and controlled. In all cases, design verification is completed before relying upon the item to perform its function.

Engineering documents (e.g., drawings, specifications, design analyses, system descriptions, engineering studies, technical reports) are verified in accordance with approved procedures and instructions.

14.6.2.4 Design Changes Design changes, including field changes, modifications to operating facilities, and disposition of nonconforming items as "use-as-is" or "repair" are justified and subject to design control measures commensurate to those applied to the original design. These control measures will include assurance that the design analyses establishing the safety basis for the SSCs are still valid. Verification and review of design changes are performed to the same level as that of the original design. As-built changes are documented and verified. Design of
temporary modifications receives the same level of control as the original design for permanent modifications. The engineering change notice is the vehicle for development, review/approval, release, and incorporation of changes to engineering documents. The engineering change notice applies to all documents released into the Project Hanford Management Contract engineering release system, except vendor information. Engineering change notices are used to:

- Revise a released engineering document
- Document temporary configuration changes
- Void or supersede an approved and released engineering document
- Cancel or supersede an outstanding engineering change notice
- Add or remove a to-be-determined or hold-point item
- Add additional sheets to an existing released drawing or multiple-sheet drawing

14.6.2.5 Design Documentation and Records. The final design is related to the design input by documentation of sufficient detail to permit design verification and by identification of assemblies and/or components that are part of the item being designed. Design documentation is adequate to support facility design, construction, and operation. Design output documents include drawings, specifications, test and inspection plans, and maintenance requirements. As-built drawings and shop drawings are maintained to show actual plant configuration. The administrative process will clearly indicate responsibilities for design output document activities including marking-up and updating during construction and operation of SNF Project activities.

14.6.2.6 Computer Software. Computer programs used for design analysis are verified to show that they produce correct solutions for the encoded mathematical model within defined limits for each parameter employed. Computer programs also are verified to show that the encoded mathematical model produces a valid solution over the range of applications to the physical problem associated with the particular application. Computer programs are controlled to ensure that changes are documented and approved by appropriate personnel. Control requirements include development, acquisition, use, modification, and configuration management of software used in computer systems.

14.6.3 Procurement

Procurement of items and services by the SNF Project or by its subcontractors is documented and controlled to ensure that regulatory requirements, design, and other necessary quality requirements are included or referenced in the documents used for procurement of items and services and that such items and services perform as specified. As stated in Section 14.3.1, suppliers and subcontractors providing services for the SNF Project are required to implement a quality assurance program consistent with the requirements specified in the SNF Project procurement documents. Quality assurance programs are submitted to the SNF Project for review and approval. Procurement documents are controlled to ensure that applicable requirements, design bases, test and inspection requirements, acceptance criteria, and other requirements necessary to ensure adequate quality are included or referenced in documents for procurement of items and services. Critical parameters and requirements are specified including...
submittal, product-related documentation, nonconformance requirements, administrative documentation, personnel or materials qualifications, tests, inspections, and reviews

A graded approach is used when establishing the stringency of procurement requirements to ensure they are commensurate with the importance of the purchased item or service to the facility or process. Applicable quality assurance requirements are applied to suppliers through the procurement documents issued to these suppliers. Changes also are communicated to suppliers in the procurement documentation. Such changes are evaluated in the same manner and use the same criteria as the original documents. The procurement documents also specify the right of access to suppliers' (including sub-tier suppliers') facilities for surveillance of work activities, inspection of facilities and programs, review of plans and program reports, processing of change information, and review of disposition of nonconformances.

Controls are established to ensure that purchased items and services conform to the procurement documents. These controls include provisions for source evaluation and selection, objective evidence of inspection at the contractor or subcontractor source, examination of items or services upon delivery and assessments. Procurement documents will require that deficiencies discovered by suppliers that involve safety-class, safety-significant, or certain general-service items and/or services are reported to the SNF Project Procurement organization. Nonconformance procedures are in place for disposition of items or services that do not meet procurement requirements.

Documentary evidence that an item conforms to code, regulation, or contract procurement requirements is completed before installation or use of the item. Methods established for the acceptance of an item furnished by suppliers may consist of one or more of the following:

- Supplier certification and release (Certificate of Conformance)
- Source verification or inspection
- Receiving inspection
- Acceptance testing
- Post-installation testing

Supplier-generated documents are controlled in accordance with the requirements in Section 14.5.

Commercial-grade items are used when practical. Requirements for commercial-grade items include specification of critical characteristics, acceptance methods specified to satisfy critical characteristics, and procurement documents identifying commercial-grade items that are to be dedicated.

An assessment to determine the effectiveness of the control of quality by the supplier is conducted, evaluated, and documented before selection and periodically during supplier performance at intervals consistent with the importance, complexity, and quantity of the items or
services The evaluation and selection of procurement sources is based on specified criteria. The evaluation will include one or more of the following:

- Evaluation of the supplier's quality history of providing an identical or similar product that performs satisfactorily in actual use.
- Review of the supplier's current qualitative and quantitative information that can be objectively evaluated.
- Direct evaluation of the supplier's facilities, personnel, and quality assurance program implementation to determine the technical and quality capability of that supplier.

Assurance is obtained that approved suppliers can continue to provide acceptable items and services based on a documented evaluation of their past performance. Suppliers are either re-evaluated and retained on the basis of continued satisfactory performance or removed from a list of acceptable suppliers. This performance history evaluation involves the following:

- Evaluations of the supplier's nonconformance report history relative to received items or services.
- Communications with the purchasing organization to determine whether any contractual problems have been encountered.
- Communications with the supplier to determine whether any changes have occurred to their quality assurance program since their initial acceptance.

The SNF Project procurement process meets the requirements of Part 2, Section 7, "Procurement," of HNF-MP-599. Implementing procedures for procurement are identified in the respective matrix associated with each SNF Project QAPP.

14.6.4 Inspection and Acceptance Testing

Inspection and acceptance testing of specified items, services, and processes are performed in accordance with established acceptance and performance criteria. Inspection and acceptance criteria are derived from engineering design documents, supplier information, construction procedures, and maintenance procedures. Inspection, surveillance, and testing of items, services, processes, operation, and interim storage that have the potential to affect quality during procurement, construction, repair, modification, maintenance, and installation are subject to these requirements on a graded approach based on safety classification.

Hold points (e.g., items or activities where inspection is mandatory), witness points, verification points, methods, acceptance criteria, checklists, and other inspection planning documents are established, documented, and implemented to ensure required inspections are performed. Test requirements and acceptance criteria are identified, documented, and approved.
Test results are documented and their conformance with acceptance criteria evaluated by a responsible authority to ensure that test requirements have been satisfied.

The inspection and testing process establishes the system by which the inspection and testing status of items is controlled to ensure that items that have not passed the required inspections and tests are not inadvertently installed, used, or operated. Status of inspection and test activities is identified either on the item by markings such as stamps, tags, labels, or routing cards, or in documents traceable to the items, including heat number, part number, serial number, or other appropriate means throughout fabrication, installation, and use.

Controls provide for the identification of items that have satisfactorily passed required inspections and tests where necessary to preclude inadvertent bypassing of the inspection and testing. Controls also identify the operating status of SSCs, such as tagging valves and switches to prevent inadvertent operation.

Inspections and tests are performed in accordance with approved procedures, instructions, or inspection plans by qualified personnel to demonstrate that the SSCs will perform satisfactorily in service. This documentation contains the following information:

- References to applicable documents such as drawings, specifications, and procedures
- Type of inspection and testing to be performed
- Mandatory hold points, when required
- Characteristics to be inspected and tested
- Qualification of individuals or groups responsible for performing the inspection and testing
- Acceptance and rejection criteria (explicit or by reference) obtained from specifications, drawings, supplier instructions, and standards
- Suitable environmental conditions
- Description of the inspection and testing method and equipment to be used, or reference to an appropriate procedure (including adequate testing instrumentation, equipment, calibration requirements, and environmental conditions)
- Required safety measures
- Frequency of inspection, testing, or sampling plan
- Results of inspections and tests
Inspection and test results are evaluated and verified by authorized personnel to document that all requirements have been satisfied. Records will identify the item examined, date of examination, examiner or data recorder, results, observations, acceptability (ensuring that all prerequisites for the given test are met), and action taken concerning any deviations noted.

Measuring and testing equipment used for inspections and tests is calibrated, maintained, and verified to be of the required precision and accuracy for use by personnel performing inspections and tests. Where possible, instruments will have calibration certifications traceable to nationally recognized standards. Instruments are calibrated at specified intervals, before and after use, or just prior to use, as determined by required accuracy, intended use, frequency of use, stability characteristics, and other conditions affecting performance. Instruments are labeled, tagged, or otherwise controlled to indicate calibration status and to ensure traceability to calibration test data. Instruments found out-of-calibration or out-of-tolerance are tagged or segregated and not used until they are successfully recalibrated. The acceptability of items or processes measured, inspected, or tested with out-of-tolerance instruments is evaluated and measurements and tests repeated as required.

The SNF Project inspection and acceptance testing process meets the requirements of Part 2, Section 8, “Inspection and Acceptance Testing” of HNF-MP-599. Implementing procedures for inspection and acceptance testing are identified in the respective matrix associated with each SNF Project QAPP.

14.6.5 Assessments

This section briefly describes performance of management self-assessments and independent assessments to determine the adequacy of the quality program. The independent assessments are performed by Fluor Daniel Hanford, Incorporated, as described in Chapter 17.

14.6.5.1 Management Assessments

Management assessments are planned, scheduled, and conducted by organization managers to regularly assess how well the organization is meeting its customer’s requirements and expectations.

Management self-assessments also include the following activities:

- Processes including planning, organizational interfaces, use of performance indicators, training and qualifications, supervisory oversight, and support
- Routinely observing personnel performing operating activities and/or interviewing workers, reviewing documentation, and conducting drills or exercises
- Assessing facility performance indicators and other operational information such as reportable occurrences, for trends in improving or deteriorating conditions
- Measuring performance based on objective standards, clearly defined goals, review, and feedback processes
Any barriers that are hindering the accomplishment of management objectives are identified, response actions documented, and corrective actions implemented.

The SNF Project management assessment process meets the requirements of Part 2, Section 9, "Management Assessments," of HNF-MP-599, and SNF Project administrative procedures. Implementing procedures for management assessments are identified in the respective matrix associated with each SNF Project QAPP.

14.6.5.2 Independent Assessments The independent assessment program (1) provides for the measurement of item and service quality, requirements compliance, and work performance and (2) promotes improvement. The assessment process is conducted using written procedures and/or checklists and incorporates a performance-based approach with emphasis on the results of work processes and compliance with requirements. Independent assessments are conducted by Fluor Daniel Hanford, Incorporated, on SNF Project activities that most directly relate to final objectives and emphasize safety, reliability, and product performance.

The type of independent assessment performed and the frequency with which an assessment is performed are based on the status, complexity, and importance of the activity or process being assessed, and the past performance of the activity or process being assessed. Independent assessments may include methods such as inspections, peer and technical reviews, audits, surveillance, and customer interviews. These independent assessments are conducted by technically qualified staff, knowledgeable in the activity or process being assessed, who have sufficient authority and autonomy from the line organizations to carry out their responsibilities. An independent assessment will include:

- Work performance and process effectiveness
- Abnormal performance and potential problems
- Improvement opportunities
- Results documentation
- Satisfactory resolution of reported problems verification

The independent assessment activity verifies by way of checking, assessing, and inspection that activities affecting the functions important to safety have been correctly performed. Persons and organizations performing these quality assurance functions report to a management level high enough to ensure that the required authority and organizational autonomy are provided, including sufficient independence from cost and schedule considerations when cost and schedule may pose conflicts with safety considerations.

Independent assessment strengths and weaknesses are documented and presented to the management of organizations responsible for performance of the subject activities or processes. Areas of weakness are then analyzed to formulate corrective actions to promote improvements. Actions are tracked and the adequacy of corrective actions, including those taken to minimize or prevent recurrence, are verified. Lessons learned are communicated to other organizations with similar activities or concerns. Follow-up action, including a subsequent assessment of the activity or process, is initiated if appropriate to determine effectiveness of the corrective action.
Facility-independent assessments also will be performed by the Facility Evaluation group using the Facility Evaluation Board. Facility Evaluation Board members are not associated with the activity or process being assessed. Responsibilities of the Facility Evaluation Board are as follows:

- Establish, implement, and maintain an effective independent oversight program to assess operations and management effectiveness across the organization in meeting established environment, safety, health, and quality assurance requirements, applicable regulations, and sound management practices.
- Provide facility managers and senior management with clear, objective, and well-documented assessments of strengths and weaknesses that affect performance to approved standards.
- Ensure that independent assessments are conducted in programmatic or functional areas of environment, safety, health, and quality assurance when required by regulation or contract requirements.

The independent assessment process as performed by Fluor Daniel Hanford, Incorporated, for the SNF Project meets the requirements of Part 2, Section 10, "Independent Assessments," of HNF-MP-599. Implementing procedures for the independent assessment process are identified in the respective matrix associated with each SNF Project QAPP.

14.7 REFERENCES


DOE O 414 1, Quality Assurance, U S Department of Energy, Washington D C


*Price-Anderson Amendments Act of 1988, 42 U S C 2210, et seq*


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Figure 14-1 Project Hanford Quality Assurance Program Document Hierarchy

This document will be cancelled upon completion of the implementation actions

DOE = U.S. Department of Energy
FDH = Fluor Daniel Hanford Inc
HASQARD = Hanford Analytical Services Quality Assurance Requirements Document
M&I = Management and Integration
OCRWM = Office of Civilian Radioactive Waste Management
PHMC = Project Hanford Management Contract
QA = quality assurance
QARD = quality assurance requirements and description
RCRA = Resource Conservation and Recovery Act
SNF = spent nuclear fuel
S/RID = standards/requirements identification document
TPA = Tn Party Agreement

November 1999
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<thead>
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<th>Acronym</th>
<th>Term</th>
</tr>
</thead>
<tbody>
<tr>
<td>BED</td>
<td>building emergency director</td>
</tr>
<tr>
<td>BEP</td>
<td>building emergency plan</td>
</tr>
<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
</tr>
<tr>
<td>DOE-HQ</td>
<td>U.S. Department of Energy-Headquarters</td>
</tr>
<tr>
<td>EAL</td>
<td>emergency action level</td>
</tr>
<tr>
<td>EDF</td>
<td>Emergency Decontamination Facility</td>
</tr>
<tr>
<td>EDO</td>
<td>Emergency Duty Officer</td>
</tr>
<tr>
<td>ENS</td>
<td>Emergency Notification System</td>
</tr>
<tr>
<td>EOC</td>
<td>Emergency Operations Center</td>
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<td>EPP</td>
<td>Emergency Preparedness Program</td>
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<td>EPZ</td>
<td>emergency planning zone</td>
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<td>ERO</td>
<td>Emergency Response Organization</td>
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<td>ERPG</td>
<td>Emergency Response Planning Guideline</td>
</tr>
<tr>
<td>FDH</td>
<td>Fluor Daniel Hanford</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>HEHF</td>
<td>Hanford Environmental Health Foundation</td>
</tr>
<tr>
<td>IC</td>
<td>incident commander</td>
</tr>
<tr>
<td>ICP</td>
<td>Incident Command Post</td>
</tr>
<tr>
<td>JIC</td>
<td>Joint Information Center</td>
</tr>
<tr>
<td>NFPA</td>
<td>National Fire Protection Association</td>
</tr>
<tr>
<td>ONC</td>
<td>Occurrence Notification Center</td>
</tr>
<tr>
<td>ORP</td>
<td>U.S. Department of Energy, Office of River Protection</td>
</tr>
<tr>
<td>PAG</td>
<td>protective action guide</td>
</tr>
<tr>
<td>PAR</td>
<td>protective action recommendation</td>
</tr>
<tr>
<td>POC</td>
<td>Patrol Operations Center</td>
</tr>
<tr>
<td>RCRA</td>
<td>Resource Conservation and Recovery Act of 1976</td>
</tr>
<tr>
<td>RL</td>
<td>U.S. Department of Energy, Richland Operations Office</td>
</tr>
<tr>
<td>SED</td>
<td>Site Emergency Director</td>
</tr>
<tr>
<td>SMT</td>
<td>Site Management Team</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
</tr>
<tr>
<td>S/RID</td>
<td>standards/requirements identification document</td>
</tr>
<tr>
<td>TEDE</td>
<td>total effective dose equivalent</td>
</tr>
<tr>
<td>UDAC</td>
<td>Unified Dose Assessment Center</td>
</tr>
<tr>
<td>WAC</td>
<td>Washington Administrative Code</td>
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</tbody>
</table>
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15.0 EMERGENCY PREPAREDNESS PROGRAM

15.1 INTRODUCTION

This chapter describes the philosophy, objectives, and organization of the Spent Nuclear Fuel (SNF) Project Emergency Preparedness Program (EPP) for response to emergencies at the SNF Project facilities. The SNF Project EPP requires the Hanford Site Emergency Response Organization (ERO) to respond to an emergency in a timely, efficient, and effective manner. The SNF Project EPP implements the requirements listed in Section 15.2 to ensure the safety and health of the workers, the public, and the environment in the event of an emergency. The EPP applies to all operations, facilities, personnel, programs, procedures, and workers associated with the SNF Project.

This chapter contains the following information:

- SNF Project commitments to meet the emergency preparedness requirements of Federal, state, and U.S. Department of Energy (DOE) rules, orders, regulations, standards, and codes listed in Section 15.2.

- Knowledge of the EPP and subsequent preparedness of the SNF Project in the event an incident should occur.

- Confirmation that the emergency preparedness readiness provided by the SNF Project EPP is appropriate and prudent and, through use of a graded approach, ensures overall consistency with the spectrum of hazards, accidents, and consequences addressed in the facility annexes to the SNF Project Final Safety Analysis Report (FSAR). The increased level of response, based on the severity of the events as called out by each of the successively higher emergency classification levels, represents a graded response embodied in the SNF Project emergency planning.

In addition, this chapter describes the integration of the EPP with DOE/RL-94-02, Hanford Emergency Management Plan, that links to the offsite emergency planning efforts. This chapter describes the features of the SNF Project EPP, the SNF Project building emergency plans (BEPs), and DOE/RL-94-02 as they relate to the SNF Project. The development of this chapter has been coordinated with the development of Chapters 10, 20, 30, 120, and 170.

15.2 REQUIREMENTS

The requirements that form the basis for the EPP are found in HNF-SD-SNF-RD-001, Spent Nuclear Fuel Project Standards/Requirements Identification Document. The specific requirements applicable to this chapter include.
15.3 SCOPE OF EMERGENCY PREPAREDNESS

Potential SNF Project emergencies span the spectrum of identified emergencies for SNF Project facilities, from worker injuries to emergencies with potential public impact. Facility events include fires and/or explosions, radioactive material releases, and hazardous material releases. Natural events include seismic events, high winds, range fires, floods, tornadoes, and aircraft crashes. Potential hazardous events that could affect SNF Project facilities are presented in Chapter 30 of the applicable facility FSAR Annex. Security events including bomb and/or explosive device threats, sabotage incidents, and hostage and/or armed intruder situations will be responded to as described in Sections 15.4.1.2 and 15.4.1.2.2.
This chapter applies to all SNF Project operations, facilities, and personnel (including subcontractors, vendors, visitors, and any noncontractor tenants).

The EPP addresses individual and organizational graded responses to the spectrum of potential SNF Project emergencies, including responses to accidents with very low occurrence frequencies. Planned responses documented in the EPP and the BEPs provide the response actions for these emergencies.

This chapter links the SNF Project EPP to DOE/RL-94-02, which provides the link to subsequent Federal, state, and local organization EPPs. Integration of these programs links potential onsite events with onsite and offsite impacts. This integration assists in mitigation and recovery and provides for protection of the health and safety of the workers, the public, and the environment (see Figure 15-1).

### 15.4 EMERGENCY PREPAREDNESS PLANNING

SNF Project emergency preparedness planning includes identification of emergency organizations, assessment actions, emergency classification notification processes, emergency facilities and equipment, protective actions, training, drills, exercises, and recovery actions. As required by Section 5.3 of the standards/requirements identification document (S/RID) (HNF-SD-SNF-RD-001), development and maintenance of emergency plans (i.e., BEPs) and emergency response procedures will be established for emergency preparedness planning.

The SNF Project facility-specific documentation will address the following topics:

1. Preemergency planning and coordination with onsite parties
2. Personnel roles, lines of authority, and communication
3. Emergency recognition and prevention
4. Safety-distances and places of refuge
5. Site security and control
6. Evacuation routes and procedures
7. Decontamination procedures
8. Emergency medical treatment and first aid
9. Emergency alerting and response procedures
10. Critique of response and follow-up
11. Protective equipment and emergency equipment

Section 5.3 of the S/RID also requires emergency response procedures and other information for handling emergency incidents, including the following:

1. Data on site topography, layout, and prevailing weather conditions
2. Procedures for reporting incidents to local, state, and Federal governmental agencies
3 Procedures for cleanup of spills or accidental releases

4 The emergency response plan shall be compatible and integrated with the disaster, fire, and/or emergency response plans of local, state, and Federal agencies

5 The emergency response plan shall be rehearsed regularly as part of the overall training program for site operations

6 The site emergency response plan shall be reviewed periodically and, as necessary, be amended to keep it current with new or changing site conditions or information

7 An employee alarm system shall be installed in accordance with Federal and state alarm requirements to notify employees of an emergency situation to stop work activities if necessary, and to begin emergency procedures

Facility-specific emergency planning requirements are implemented through administration of the SNF Project EPP. The accident categories derived from each of the SNF Project facility emergency planning hazards assessments drive the planning process. Input for the specific facility emergency planning requirements is obtained from Chapter 30 of the applicable facility FSAR Annex. As modifications to SNF Project facilities are planned during design, construction, and operation, the hazards surveys and hazards assessments will be reevaluated to reflect changes. Subsequent changes to the SNF Project EPP may be required.

Hanford Site-specific emergency planning requirements are implemented through DOE/RL-94-02, which applies to all Hanford Site employees and facilities. Hazardous material emergencies are grouped into one of three classes (Alert, Site Area Emergency, or General Emergency) according to magnitude or severity. This method of classification promotes timely and effective response by triggering planned response actions appropriate to all events of a given class. The increased level of response, based on the severity of the event, as called out by each of the successively higher emergency classification levels, constitutes a graded approach.

15.4.1 Emergency Response Organization

The mission of the Hanford Site ERO is to ensure that, in the event of an emergency, actions will be taken to prevent or minimize impacts to workers, the public, facilities, and the environment. As stated in requirements of Sections 5.1 and 5.2 of the S/RID, the Hanford Site ERO is structured and staffed with trained personnel, including designated alternates, to enable the most timely and effective response possible while meeting the requirements set forth in DOE O 151.1, WAC 173-303 and other applicable Federal and state regulations identified in the S/RID, including Hazardous Waste Operations and Emergency Response, Occupational Safety and Health Administration and National Fire Protection Association (NFPA).
15.4.1.1 Organization Structure  Emergency response on the Hanford Site will be performed in accordance with the Incident Command System, which is an integrated emergency management system with clearly defined responsibilities and communication pathways that allows predesignated trained individuals to jointly determine and implement incident mitigation strategies. While the U.S. Department of Energy, Richland Operations Office (RL) maintains the option to assume overall management, direction, and control of any Hanford Site emergency, the Hanford Site ERO has been formed to allow the Hanford Site contractors to continue their management and operational roles in the event of an emergency. Specific DOE Office of River Protection (ORP) responsibilities within the Hanford Site ERO are presently being defined as part of the ongoing RL/ORP interface clarification discussions.

The Hanford Site ERO has two components. The first component, the Incident Command Organization, consists of the SNF Project Facility ERO and Site contractor emergency response personnel (i.e., Hanford Fire Department, Hanford Patrol). The Facility ERO has responsibility for implementing emergency response activities at the SNF Project facility involved in the event. The site contractor emergency personnel have responsibility for on-scene mitigation of the event.

The second component, the DOE Hanford Emergency Operations Center (EOC), has the responsibility to monitor and provide support for onsite response, assist with issue resolution, assess the offsite impacts, and interface with offsite agencies and the public. The Incident Command System at the Hanford Site is shown in Figure 15-2, the emergency communication chain is shown in Figure 15-3.

A graded approach is used to respond to an event depending upon the nature of the facility and/or the severity of the event. There are a number of events to which the SNF Project facilities have to be ready to respond, including hazardous material releases, spills, operational events, fires, natural phenomena, and security events.

15.4.1.2 Incident Command Organization  As stated above, the Incident Command Organization is composed of two main groups: the Facility ERO and Site contractor emergency response personnel (i.e., Hanford Fire Department and Hanford Patrol). The Building Emergency Director (BED) is responsible for implementing facility response procedures (e.g., protective actions, event classification, notification) until arrival of the Hanford Fire Department or Hanford Patrol Incident Commander (IC). The IC is responsible for the health and safety of all personnel at the event scene.

Upon arrival of the Hanford Fire Department or Hanford Patrol IC, the Facility ERO becomes part of a consolidated Incident Command Organization. The senior Hanford Fire Department official becomes the IC unless the event is primarily a security event in which case the Hanford Patrol senior officer becomes the IC. If the BED is not present at the event facility at the start of the event, the IC will perform the duties of the BED in addition to the IC duties pending arrival of the BED. However, in accordance with DOE/RL-94-02, the BED (primary or alternate) will be on the facility premises in control of the work during any work with hazardous operations that could result in generating an Alert or higher emergency. On-call BEDs will be used only for facilities where hazardous materials are in storage and stable and the only work...
being performed is that of surveillance. SNF Project Operations will maintain a listing of on-call BEDs assigned to non-hazardous and hazardous facilities with work and home telephone numbers at the Occurrence Notification Center (ONC).

The BED retains responsibility for direct configuration control over SNF Project facility systems and components, the IC controls the overall management strategy associated with the incident and ensures that all functional areas are staffed and working effectively to mitigate the incident. At the site level, the Hanford Patrol security services provides services such as activation of crash alarm telephone systems and sirens, coordination of the movement of emergency personnel through security gates, evacuation assistance, and barricade establishment where needed. Additional law enforcement is available from federal and local agencies at the request of RL/ORP.

The Incident Command Organization has the authority to commit resources (equipment and personnel) needed to carry out the emergency response and is responsible for being thoroughly familiar with applicable plans and procedures, operations and activities at the facility, location and properties of all wastes handled, location of all records within the facility, and the layout of the facility. The Incident Command Organization will be staffed by pre-appointed and trained personnel. Initial, annual, and refresher training of the Incident Command Organization personnel on their respective roles, responsibilities, and authorities is presented in Section 15.4.7. Table 15-1 provides a summary of all Incident Command Organization functions and the staffing responsibilities for each function.

<table>
<thead>
<tr>
<th>Function</th>
<th>Responsible Staffing</th>
</tr>
</thead>
<tbody>
<tr>
<td>Incident Commander</td>
<td>Hanford Fire Department or Hanford Patrol</td>
</tr>
<tr>
<td>Building Emergency Director</td>
<td>Affected facility</td>
</tr>
<tr>
<td>Public Information Officer</td>
<td>FDH or appropriate contractor personnel</td>
</tr>
<tr>
<td>Liaison Officer</td>
<td>Emergency Duty Officer</td>
</tr>
<tr>
<td>Safety Officer</td>
<td>Hanford Fire Department</td>
</tr>
<tr>
<td>ICP Communicator</td>
<td>Affected hazardous facility</td>
</tr>
<tr>
<td>ICP Hazards Communicator</td>
<td>Affected hazardous facility</td>
</tr>
<tr>
<td>Facility Operations Specialists</td>
<td>Affected facility</td>
</tr>
<tr>
<td>Operations Section Chief</td>
<td>Hanford Fire Department or Hanford Patrol</td>
</tr>
<tr>
<td>Security</td>
<td>Hanford Patrol</td>
</tr>
<tr>
<td>Radiological Hazards Assessor</td>
<td>Affected facility radiological control manager (or equivalent)</td>
</tr>
<tr>
<td>Chemical Hazards Assessor</td>
<td>Hanford Fire Department on call Industrial Hygienist or affected facility</td>
</tr>
<tr>
<td>Communications Specialist</td>
<td>FDH</td>
</tr>
</tbody>
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Table 15-1 Incident Command Organization Functions (2 sheets)

<table>
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<tr>
<th>Function</th>
<th>Responsible Staffing</th>
</tr>
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<tbody>
<tr>
<td>Planning Section Chief</td>
<td>Hanford Fire Department</td>
</tr>
<tr>
<td>Logistics Section Chief</td>
<td>Hanford Fire Department</td>
</tr>
<tr>
<td>Resource Staging Area Manager</td>
<td>Hanford Fire Department</td>
</tr>
<tr>
<td>Facility Staging Area Manager</td>
<td>Affected facility</td>
</tr>
</tbody>
</table>

FDH = Fluor Daniel Hanford  
ICP = Incident Command Post

The Hanford Fire Department senior officer will be the IC for events involving fire, medical, hazardous materials, or rescue and also fill the role of the senior emergency response official. The Incident Command Organization, however, may be staffed entirely with SNF Project personnel for less significant events as determined by the BED. Events that might not require assistance from outside the SNF Project facility would include small releases of known substances when mitigation can be accomplished by trained on-scene personnel, minor first aid cases, non-injury contamination incidents, and non-emergency plant responses. In this instance, the BED coordinates emergency response efforts at the scene including oversight of mitigation efforts, use of appropriate personal protective equipment, facility protective actions, and notifications.

An Incident Command Post (ICP) will be established in a safe location near the event scene where emergency responders from different organizations may collocate to coordinate actions. Organizations supporting the ICP retain responsibility for their technical expertise and provide facility expertise to the ICP.

15 4 1 2 1 Spent Nuclear Fuel Project Emergency Response Organization Initial direction and control of emergency response at an SNF Project facility is the responsibility of the Facility ERO. The BED directs the Facility ERO who will assist in the protection of personnel property, and the environment. SNF Project senior management will designate the BEDs. This organization (Facility ERO) will direct appropriate emergency response actions at the event scene prior to arrival of the IC, including effective coordination with the DOE Hanford EOC.

Collective responsibilities of SNF Project senior management, BED and other Facility ERO members for hazardous facilities include the following, as applicable:

1. Assigning and assuring the training of the Facility ERO (BEDs, personnel accountability aides, staging area managers and other facility personnel as appropriate) to support the Hanford Fire Department as the primary hazardous materials emergency response agency.

2. Maintaining, reviewing, revising, and implementing the BEP and applicable facility-specific emergency response procedures.
Ensuring that facility personnel are aware of hazards in and around the SNF Project

Ensuring that facility personnel are trained to respond to emergencies

Determining when an event has occurred or a condition exists that requires appropriate emergency event classification in accordance with DOE/RL-94-02 and DOE O 232 1A

Assisting in emergency classification and emergency notification of such classification within established regulatory time limits

Contacting the Patrol Operations Center (POC) to implement predetermined onsite protective actions and provide initial emergency and classification information in accordance with established procedures

Reporting events or conditions in accordance with applicable federal and state regulations

Assisting in alerting employees of an emergency situation

Activating internal facility alarms or communications systems, where applicable, to implement actions to protect workers within their respective geographic area of responsibility as defined in the BEP or procedures

Assisting in the safe evacuation of the incident scene hazard area, to include emergency lifesaving rescue if required

Providing immediate first-aid if required

Placing operating systems or controls in a safe configuration

Providing event status information to the DOE Hanford EOC

Providing personnel to staff the DOE Hanford EOC including senior management staff and technical representation

Assessing the potential or actual onsite and offsite consequences of the emergency

Implementing or supporting the implementation of protective actions for the general population to include roadblocks and building sweeps

Providing assistance to the Hanford Fire Department to include meeting and directing responders to the event scene, providing safe routes of travel, and providing immediate and constant interface, coordination, and information as the emergency situation requires
Assisting the IC, as necessary, in the mitigation of emergencies within the assigned building by

- Identifying the character, exact source, amount, and extent of any released materials
- Assessing possible hazards to human health and the environment that may result from a release, fire, or explosion
- Taking reasonable measures (e.g., stopping processes and operations, collecting and containing released waste, removing or isolating containers) necessary to ensure that fires, explosions, and releases do not occur, recur, or spread to other dangerous waste
- Monitoring for leaks, pressure buildup, gas generation, or ruptures in valves, pipes, or other equipment, as appropriate

Ensuring that building occupants take appropriate protective actions in response to events occurring in other onsite geographic areas or adjacent facilities

Establishing an initial ICP and assigning other Incident Command Organization functions in accordance with established procedures to provide effective control at the event scene

Serving as emergency response team members in support of the Hanford Fire Department for entry into the incident scene hazard area for mitigation where personnel protective equipment requirements do not specify Level A or Level B dermal protection

Providing radiological monitoring and assessment for emergency response

Providing support for chemical and/or radiological decontamination

Ensuring proper cleanup, transportation, and storage of hazardous materials generated as a result of the event

Providing funding for performance of emergency response and recovery duties and replacement of supplies used by other contractors for event response

Responsibilities of the Facility ERO for administrative and non-hazardous facilities are contained in Section 2 2 1 1 of DOE/RL-94-02. As required by the S/RID (HNF-SD-SNF-RD-001), the positions and responsibilities of the Facility ERO including alternates, will be contained in the facility-specific BEP.
In addition to the BED, the Facility ERO will consist of trained support staff (e.g., Health Physics). In addition, personnel accountability aides are responsible for facilitating the implementation of protective actions (evacuation or take cover) and for facilitating the accountability of personnel after the protective actions have been implemented. Staging area managers are responsible for coordinating and conducting activities at the staging area. Personnel accountability aides assist the staging area managers by ensuring that personnel and visitors are properly evacuated from designated staging areas to a safe location. The Facility ERO supports actions requested by the IC.

15 4 1 2 2 Site Contractor Emergency Response Personnel

15 4 1 2 2 1 Hanford Fire Department As stated in Section 15 4 1 2, the Hanford Fire Department is the designated hazardous material incident command agency responsible for control of all hazardous material and chemical or biological incidents on the Hanford Site, including fire control, releases of hazardous materials, and personnel rescue response activities associated with the incident. The Hanford Fire Department provides hazardous materials first responders as defined in 29 CFR 1910 120(q)(6)(i)-(v) and NFPA 472, and a qualified Safety Officer for all emergency response activities.

The Hanford Fire Department also monitors facility fire alarm systems and coordinates and provides emergency medical services on the Hanford Site as a 24-hour operational facility and dispatch center.

15 4 1 2 2 2 Hanford Patrol The Hanford Patrol serves as the security agency for the Hanford Site and provides services such as coordinating the movement of emergency responders through security gates, assisting evacuation, and establishing barricades when needed. As stated in Section 15 4 1 2, the Hanford Patrol is the IC for security emergencies.

The POC, a 24-hour operational facility, is responsible for emergency functions such as the following:

- Acting as the single point-of-contact to initiate emergency response by
  - Operating the Site's enhanced 911 system
  - Notifying the BED (when not on the premises)
  - Requesting response from the Hanford Fire Department
  - Contacting Hanford Environmental Health Foundation (HEHF) on-call provider for medical incidents involving radiological or chemical exposures
  - Notifying other appropriate on-call personnel
  - Activating or requesting activation of appropriate alarm signals

November 1999
Providing information of onsite medical emergencies to appropriate contractor organizations

- Activating the ONC conference bridge upon notification of a declared emergency and implementing onsite protective actions by activating or requiring activation of warning sirens and crash alarm telephone systems

- Receiving emergency response telephone calls during offsite shipments of RL/ORP-owned hazardous materials

15 4 1 2 3 Hanford Environmental Health Foundation Services provided by HEHF in an emergency include the following

- Minor emergency medical care and consultation

- Medical support for chemical and radiologically contaminated patients

- Medical staffing and operation of the Emergency Decontamination Facility (EDF) and Health Care Centers

- Hostage negotiation and critical stress debriefing support

- Coordination with and support to community medical services

- Senior management and technical staff support to the DOE Hanford EOC

15 4 1 2 3 Other Emergency Response Support Personnel Some emergency situations may require facility or site support personnel for emergency response at the event scene that are not assigned permanent positions within the Hanford Site ERO These emergency response support personnel — termed either as Skilled Support Personnel or Specialist Employees — are not trained to operate within the Hanford Incident Command System and will be used for specific tasks defined in the following subsections

15 4 1 2 3 1 Skilled Support Personnel Personnel needed to operate specific support equipment, including those within the incident scene hazard area but who are not addressed in specific emergency response procedures, may be designated as Skilled Support Personnel Such personnel receive a briefing prior to commencing any work

15 4 1 2 3 2 Specialist Employees Safety professionals and environmental specialists who provide technical advice within their field of expertise but who are not addressed in specific emergency response procedures, may be designated as Specialist Employees Such personnel provide expertise and advise to the IC when requested, however they may not enter the incident scene hazard area
Department of Energy Hanford Emergency Operations Center

The DOE Hanford EOC is an emergency response facility maintained by RL/ORP for the purpose of providing an area where personnel may convene during emergency conditions to provide essential response functions. These functions include public information, offsite protective action recommendations (PARs), field monitoring and sampling, hazards assessments, oversight of onsite mitigative activities, and oversight of onsite protective actions.

The DOE Hanford EOC is activated upon declaration of an Alert or higher emergency. The DOE Hanford EOC also may be fully or partially activated under the following circumstances:

- As directed by the RL/ORP manager or designee, when an event occurs that is not classified as an Alert or higher emergency but where action to provide monitoring assistance to the event scene or other agencies is requested, such events may include the following:
  - Hanford Site emergency conditions that potentially involve significant onsite or offsite consequences
  - Security events
  - Natural disasters (e.g., earthquake, tornado) that could or do result in significant onsite or offsite public or environmental impact
  - Requests from other government agencies for support of regional emergencies
  - Threats or acts of terrorism, or when a national emergency is declared by the President of the United States or the United States Congress

- As directed by the Radiological Assistance Program team leader to support a Radiological Assistance Program response

- In response to non-DOE emergencies that affect the Hanford Site

- In response to transportation emergency preparedness events involving the offsite shipment of DOE-owned hazardous materials

The DOE Hanford EOC is made up of several organizations that are responsible for implementing defined emergency response tasks. These organizational areas are defined in the following subsections. Detailed procedures for the activation, staffing, and operation of the DOE Hanford EOC are contained in DOE-0223, *U.S. Department of Energy Richland Operations Office Emergency Plan Implementing Procedures*. Any emergency procedures applicable to SNF Project facilities that interface with the DOE Hanford EOC procedures are included in the facility-specific procedure manual.
15 4 1 3 1 Policy Team  The primary functions of the Policy Team are the oversight of onsite activities, approval and communication of offsite PARs, approval of reclassification recommendations, oversight of public information activities, and coordination with offsite agencies. The Policy Team is staffed by the RL/ORP Emergency Manager, Public Information Director, Emergency Preparedness Advisor, Offsite Interface Coordinator, DOE Headquarters Liaison, and the responding state and county representatives.

During security events, RL is responsible for all decisions that address mitigation of the security event. This involves direction and control of Hanford Site security and patrol forces and coordination of facility response. However, the Federal Bureau of Investigation may exercise the option to take command of security events involving the violation of the Atomic Energy Act of 1954 or other federal statutes. Associated response by site contractor personnel for personnel and operational safety rests with the IC and the BED.

The Policy Team performs the following duties:

- Overview of onsite response and mitigation actions and assistance to the SNF Project staff.
- Offsite notifications and PARs to state, local, and Federal agencies and continuous updates to the state and counties about conditions.
- Notification to the DOE-Headquarters (DOE-HQ) if facility operations are shut down as a part of the protective action response.
- Direction and control, as appropriate, during a security incident.
- Reclassification or termination of the emergency.
- Direction of the activities of the Joint Information Center (JIC) in providing timely and accurate release of information to the public and media, including approval of all RL/ORP news releases.
- Requests for additional DOE emergency response assets to the Regional Response Coordinator as needed.
- Liaisons to offsite emergency centers and responding DOE emergency response assets.
- Representative to DOE-HQ if requested.
- Designation of a recovery organization.

15 4 1 3 2 Joint Information Center  The primary function of the JIC is the dissemination of accurate and timely information to the public and employees about RL/ORP activities during declared emergencies. The JIC is staffed by RL/ORP, contractor state and county representatives.
communication professionals responsible for coordinating the release of information to the public and media.

Press releases are reviewed for technical accuracy and security sensitivities prior to approval by the RL Public Information Director. Upon approval, the press releases are sent to the JIC for dissemination.

The JIC provides a single location where RL/ORP and Hanford Site contractors can coordinate the release of information with other federal agencies, state, and local jurisdictions. The JIC operates under the direction of the RL Public Information Director and is staffed by RL/ORP and Site contractor personnel. Provisions will be made at the JIC for representatives from the states of Washington and Oregon, plume emergency planning zone (EPZ) counties, and other federal agencies that may be involved in the emergency response.

The following functions are performed at the JIC:

- Preparing and coordinating information released to the public and media
- Answering questions of the public and media
- Rumor control

**15 4 1 3 3 Site Management Team** The primary functions of the Site Management Team (SMT) are to provide support to the Incident Command Organization by providing additional resources not easily obtained by the IC, tracking the status of onsite protective actions and directing implementation of additional onsite protective actions as required, and providing communications support. The SMT also is responsible for hazards assessments activities, tracking personnel medical issues, developing additional offsite PARs, and record keeping. The SMT is made up of four support organizations that are responsible for implementing defined emergency response tasks. These organizations are defined below:

**15 4 1 3 3 1 Executive Team and Support Staff** The Site Emergency Director (SED) is responsible for the coordination of all SMT activities. In this role, the SED is responsible for the activities of the Event Support Coordinator, EOC Operations Manager, and the Consequence Assessment Director. Because RL has an operational function over Hanford Site security forces, the Security Director is responsible for the activities of the Security Operations Coordinator. The Security Director will receive information from and provide direction to the security forces. The Security Director will communicate planned actions of security forces to the SED and Safety Oversight Director to ensure all safety and security issues are addressed and coordinated. The SED, in conjunction with the Security Director and Safety Oversight Director, is responsible for periodically providing status information to the RL/ORP Emergency Manager and the Policy Team. The contractor representative and SMT Emergency Preparedness Advisor provide support to the SED. In this role, the SED reports to the RL/ORP Emergency Manager.
15 4 1 3 3 2 Security and Event Support  As part of the SMT staff, the Security Operations Coordinator's primary functions are security operations, which include interface with local law enforcement agencies, coordination with the Federal Bureau of Investigation, and oversight of onsite patrol activities. In this role, the Security Operation Coordinator reports directly to the Security Director.

The Event Support Coordinator is responsible for event support activities, which include Hanford Site support services, technical support, communications with the event scene, and coordination with the EDF and other medical assessment activities. In this role, the event support coordinator reports directly to the SED.

15 4 1 3 3 3 Unified Dose Assessment Center  As part of the SMT, the primary Unified Dose Assessment Center (UDAC) functions are monitoring and evaluating existing emergency conditions to develop additional PARs. The UDAC is responsible for field team activities which include plume tracking, monitoring, and sampling.

Representatives from the states of Washington and Oregon participate in the development of recommendations and provide direction for offsite environmental monitoring. The UDAC is operated by Hanford Site contractor personnel with knowledge in the technical areas of meteorology, toxicology, industrial hygiene, and health physics. The Consequence Assessment Director is responsible for all UDAC activities. In this role, the Consequence Assessment Director reports directly to the SED.

Specific UDAC responsibilities include:

- Acquiring necessary data and measurements to evaluate personnel radiation doses and chemical exposures resulting from the event.
- Assessing the potential for onsite and offsite consequences of a release of radioactive or nonradioactive materials based on meteorological conditions, source term, location, and dispersal of the hazardous material.
- Assisting the SNF Project Staff or other Hanford Site contractors in onsite hazards assessments or development of onsite protective actions.
- Analyzing the consequences associated with evacuating versus remaining in a take cover situation for onsite personnel and recommending approved additional protective actions if necessary.
- Developing offsite PARs in coordination with representatives from the states of Washington and Oregon.
- Coordinating and directing emergency environmental monitoring teams that are not assigned to the event facility, this may include state field teams performing offsite monitoring if requested by the states.
15 4 1 3 3 4 Department of Energy Hanford Emergency Operations Center Operations

As part of the SMT, the primary functions of the DOE Hanford EOC Operations team are administration, record keeping, and dissemination of information to offsite agencies (e.g., RL Notification Form, UDAC products). The EOC Operations Manager is responsible for these activities. In this role, the EOC Operations Manager reports directly to the SED.

15 4 1 4 Offsite Emergency Support Resources

RL interfaces with federal, state, tribal, and local organizations responsible for offsite emergency response and protection of the public and environment. These organizations work with RL in developing integrated programs and response plans for emergencies at the Hanford Site. Typical interfaces are with state and local emergency management organizations, local hospitals, law enforcement agencies, emergency medical service providers, and fire departments. Coordination with offsite agencies includes planning for potential offsite protective actions. These actions may include controlling access to State Highway 240 and the portion of the Columbia River adjacent to the Hanford Site, and providing protective action recommendations for the public in other offsite areas that may be impacted by an emergency.

RL Security and Emergency Services and the Hanford Fire Department execute and maintain memoranda of understanding with various offsite entities related to emergency preparedness assistance. Copies of the memoranda of understanding are provided in DOE/RL-94-02, Appendix B.

15 4 2 Assessment Actions

Hazards surveys and hazards assessments will be developed and maintained for emergency planning purposes. This section summarizes the process by which the onset of an emergency is recognized and the methodology used to obtain meteorological information and estimate release rates and source terms. The computer models used in the UDAC are summarized. SNF Project personnel use approved and validated emergency response guides that prescribe personnel actions during a facility emergency.

The hazards surveys will briefly describe the potential impacts of SNF Project facility emergency events or conditions and will summarize the planning and preparedness requirements that will apply. The hazards assessments will include the identification and characterization of hazardous materials (radiological and non-radiological) specific to a SNF Project facility or activity analyses of potential accidents or events, and evaluation of potential consequences.

Initial and continuous consequence assessments will be performed to protect workers, the public, and the environment during a declared emergency (See Section 15 4 2 3). Consequence assessments will evaluate and interpret radiological, other hazardous materials measurements, or other information to provide a basis for decision-making. This planning will include developing and preparing postulated scenarios for onsite and offsite consequence projections for development of PARs and identifying personnel and resources to provide an effective response.
15 4 2 1 Hazards Surveys  DOE O 151 1 requires that emergency management efforts begin with the identification of hazards and that the scope and extent of emergency planning and preparedness be commensurate with these hazards. Hazards surveys will be prepared for each SNF Project facility to identify the conditions to be contained in the comprehensive emergency management program. The hazards surveys will:

- Identify and describe the facility or activity
- Describe potential impacts of events or conditions
- Describe the potential health, safety or environmental impacts related to specific and nearby facilities
- Summarize the planning and preparedness requirements that apply

The hazards surveys will be updated as changes warrant a change but not less than every three years.

15 4 2 2 Hazards Assessments  DOE O 151 1 requires a facility-specific emergency planning hazards assessment for each facility that has the potential to generate an Alert or higher emergency. The hazards assessments are prepared from the facility-specific hazard and safety analyses that are developed and contained in Chapters 3 0 of the facility FSAR Annexes. Hazards assessments are also derived from other pertinent facility documentation (e.g., safety assessment documents, interim safety basis documents, and special nuclear material accountability documents). The hazards assessment provides the technical basis for the emergency management program. The scope and extent of planning and preparedness directly corresponds to the type and scope of hazards present and the potential consequences of events.

The observable methods of detecting or recognizing an emergency will be identified using the accident scenarios and consequences contained in SNF Project facility hazards assessments. Indicators and emergency action levels (EALs) described in Section 15 4 3 5 will be used to determine the emergency class. The emergency class will trigger specified, preplanned responses and protective actions. For each emergency class, predetermined protective actions necessary to protect onsite personnel and recommended actions for protection of offsite personnel will be established.

The hazards assessments characterize the potential consequences on workers, the public, and the environment for each postulated accident and determine the EPZ for each facility as well as the emergency class protective actions, and the observable indications and criteria (EALs) corresponding to the range of identified accidents.
A spectrum of potential accidents ranging from minor to beyond design basis are postulated and realistically analyzed. While not every conceivable situation will be analyzed, the hazards assessments will provide the framework for response to virtually any declared emergency.

Hazards assessments will be reviewed at least annually and updated, as necessary, whenever the facility configuration changes, source terms change, or the operations of the facility are modified, and are maintained in accordance with document control requirements of Section 15.4.2.3.

**15 4 2 3 Consequence Assessments** Section 5.6.1 of the S/RID (HNF-SD-SNF-RD-001) requires consequence assessments to be performed. The accident analysis in the emergency planning hazards assessment provides a starting point for real-time consequence assessments during an SNF Project emergency. These accident scenarios are summarized in DOE-0225 and are intended for use in the DOE Hanford EOC. DOE-0225 includes source terms for each analyzed accident scenario and results for adverse meteorology. The usual sequence of events after DOE Hanford EOC activation is to select a preanalyzed accident scenario that is similar to the actual event. The source term for the scenario is used with real-time meteorology to predict the potential consequences. These consequences and facility conditions form the basis for the PARs. As more information becomes available from the event scene, the source term will be refined using improved knowledge of the actual material at risk, facility damage, and standard references for release fractions.

Consequence assessment activities will:

- Be timely throughout the emergency
- Be integrated with the event classification
- Incorporate monitoring of specific indicators and field measurements
- Be coordinated with offsite agencies

The airborne release pathway typically represents the most time-urgent situation, requiring a rapid, coordinated response.

**15 4 2 3 1 Meteorological Monitoring** The routine collection of meteorological data is currently required to support environmental monitoring activities on the Hanford Site. Characterization of atmospheric transport and diffusion conditions (e.g., wind speed, wind direction, stability) in the vicinity of SNF Project facilities is essential for consequence assessments of airborne releases of hazardous materials. Weather information is updated every 15 minutes.

**15 4 2 3 2 Water/Groundwater Monitoring** As stated above, releases to aquatic and ground pathways may not have the same time-urgency as airborne releases. However, consideration of these pathways is part of the consequence assessment activities for the SNF.
Project  The water and groundwater monitoring and environmental surveillance programs required by DOE Order 5400 1 are used to characterize transport and diffusion of accidental releases of hazardous materials to aquatic pathways in the vicinity of an SNF Project facility.

15 4 2 3 3 Event Scene Consequence Assessments  Assessments will be conducted at the event scene by the Incident Command Organization staff. The Incident Command Organization staff will continuously evaluate the environmental conditions for inhabitants of the ICP and relocate the ICP as necessary.

15 4 2 3 4 Area Consequence Assessments  Consequences of releases of radioactive and non-radioactive materials at locations beyond the immediate vicinity of the event scene (e.g., 100N, 200E, 200W, 300, 400 Area) will be evaluated. These evaluations will determine the ability of operations staff to safely shutdown operational facilities and those that affect the ability of residents to take protective actions. This activity of determining impacts to other Hanford Site populations is typically performed by UDAC as indicated below and in Section 15 4 1 3 3 3.

15 4 2 3 5 Coordination of Consequence Assessment Results  The UDAC has the primary responsibility for overall onsite and offsite consequence assessments for the SNF Project and the Hanford Site. UDAC will calculate the consequences of any release of hazardous materials during an emergency. The results of these calculations will be shared with onsite and offsite emergency responders and appropriate protective action decisions and recommendations will be disseminated to affected individuals. RL will provide for representatives of the states of Washington and Oregon to participate in the consequence assessments, field team coordination and the offsite PAR development process.

15 4 2 3 6 Computer Models Used For Consequence Assessments  Two primary calculation models will be used to calculate SNF Project facility emergency airborne releases for consequence assessments. HUDU (Scherpelz 1991) and EPIcode (Homann 1988) HUDU calculates radiological releases by using a straight-line Gaussian plume and Pasquill-Gifford stability classes. EPIcode calculates hazardous chemical releases by using plume and puff Gaussian dispersion modeling, depending on the duration of the release. Other codes are available in the emergency center for specialized applications or more refined analysis.

Model parameters used in the preanalyzed hazards assessment scenarios include the default facility boundary distance of 100 m from the identified release point. Determination of release information is based on assumptions concerning the source term, release duration, wind direction, and wind speed. For example, in modeling ground-level releases, an F-stability class, 1 m/sec wind speed, a 60-m mixing level height, and a boundary receptor distance of 100 m is assumed. Elevated releases are modeled with a stack height of 61 m in accordance with HNF-IP-1201, Guidance for Conducting Emergency Preparedness Hazards Assessments. Consequence assessments during an event will use actual meteorological conditions.
15 4.3 Event Categorization and Classification

Section 5.5.1 of the S/RID (HNF-SD-SNF-RD-001) presents requirements for event classification and EALs. Five event categories will be used to meet DOE Orders, state and Federal regulators, and mutual agreements between RL and state and county agencies. These include Operational Emergencies, Resource Conservation and Recovery Act of 1976 (RCRA) Emergencies, Abnormal Events, Unusual Occurrences, and Off-Normal Occurrences. Provisions described below will be established and maintained to recognize, categorize, and classify events in order to protect workers, the public, and the environment. The Unusual and Off-normal Occurrence categories are used solely for occurrence reporting purposes.

15 4.3.1 Operational Emergency. Operational Emergencies are unplanned, significant events or conditions that require time-urgent response from outside the immediate or affected facility or area of the incident. Operational Emergencies consist of Base Program Operational Emergencies or Hazardous Material Operational Emergencies. Incidents that do not pose a significant hazard to safety, health, and/or the environment and that do not require a time-urgent response are not Operational Emergencies. Procedures to ensure recognition and appropriate categorization and classification of emergencies will be established and maintained.

15 4.3.1.1 Base Program Operational Emergency. Events may occur that represent a significant degradation in the level of safety at an SNF Project facility and that require time-urgent response efforts from outside the facility but do not involve the release or potential release of significant quantities of radiological or nonradiological materials. Such programmatic emergencies do not require classification (i.e., as Alert, Site Area Emergency, or General Emergency). The designated point-of-contact (e.g., BED, contractor single point-of-contact) with assistance from ONC personnel will assess event information to determine whether the event should be categorized as a Base Program Operational Emergency. The criteria for categorization of a Base Program Operational Emergency are part of the Abnormal Event criteria, which are contained as a single criteria list within Hanford implementing directive DOE/HFID 232 1B Notification Reporting and Processing of Operations Information. The ONC will determine whether the event should be categorized as a Base Program Operational Emergency in accordance with the criteria in DOE 0 232 1A, Occurrence Reporting and Processing of Operations Information. Base Program Operational Emergency notification requirements are provided in Section 15 4.4.1.1.

15 4.3.1.2 Hazardous Material Operational Emergency. If an Operational Emergency represents a specific threat to workers and the public because of the release or potential release of significant quantities of radiological and nonradiological hazardous materials, it will be classified as either an Alert, Site Area Emergency, or General Emergency, in order of increasing severity. See the following sections and Tables 15-2 and 15-3 for specific criteria. Periodically the emergency classification will be reviewed to ensure that the classification is commensurate with response activities. The classification will not be downgraded, however, until termination of the event. Hazardous Material Operational Emergency notification requirements are provided in Section 15 4.4.1.2.
Alert

As stated in DOE/RL-94-02, an Alert will be declared when events are predicted, in progress, or have occurred that involve an actual facility or the potential substantial degradation of the level of control over hazardous material (radiological and nonradiological). An Alert will be declared when the entire Hanford Site ERO is required to provide more than event monitoring or minimal assistance to the facility organization. An Alert will be declared for release of radiological material with a potential dose at the facility boundary greater than or equal to 100 mrem total effective dose equivalent (TEDE). Emergency action levels for hazardous nonradiological material are based on Emergency Response Planning Guidelines (ERPGs) (AIHA 1988). Alert levels for hazardous nonradiological material will be potential exposure to air concentrations of greater than or equal to ERPG-1 but less than ERPG-2 at the facility boundary (See Tables 15-2 and 15-3).

### Table 15-2: Hanford Site Hazardous Material Operational Emergency Classification Criteria

<table>
<thead>
<tr>
<th>Alert</th>
<th>Site Area Emergency</th>
<th>General Emergency</th>
</tr>
</thead>
<tbody>
<tr>
<td>&gt;ERPG-1 and &lt;ERPG-2 at facility boundary</td>
<td>&gt;ERPG-2 at facility boundary</td>
<td>&gt;ERPG-2 at Hanford Site boundary</td>
</tr>
<tr>
<td>≥100 mrem TEDE at facility boundary</td>
<td>≥1 rem TEDE at facility boundary</td>
<td>≥1 rem TEDE at Hanford Site boundary</td>
</tr>
</tbody>
</table>

Appropriate ERPG values or equivalent as stated in the hazards assessment guidance document. Solubility class D uranium compounds are limited by chemical toxicity.

*Facility boundary is defined as the properly protected area perimeter fence when present or a distance of 200 m from the release location unless otherwise specified in the hazards assessments documentation.

*The TEDE includes the summation of the doses delivered from plume submersion, ground shine, and inhalation from accidental releases.

ERPG = Emergency Response Planning Guidelines
TEDE = total effective dose equivalent
Table 15-3  Summary of Hazardous Material Operational Emergency Classifications

<table>
<thead>
<tr>
<th>Operational Emergency Classification</th>
<th>Facility or Process Event</th>
<th>Safeguards and Security</th>
<th>Onsite Transportation Event</th>
</tr>
</thead>
<tbody>
<tr>
<td>Alert</td>
<td>Actual or potential substantial degradation of level of control over radiological or nonradiological hazardous material. Releases are not expected to exceed applicable PAG or ERPG levels at facility boundary OR Actual or potential substantial degradation in the level of safety or security that could with further degradation produce a Site Area Emergency or General Emergency</td>
<td>Substantial actual or potential degradation of level of protection that could with further degradation, lead to the loss of special nuclear material or a radiological sabotage event</td>
<td>Actual or potential substantial degradation of the safety of the shipment. Exposures in excess of PAG or ERPG levels only expected for personnel engaged in cleanup recovery and investigation</td>
</tr>
<tr>
<td>Site Area Emergency</td>
<td>Actual or potential major failures of functions necessary for the protection of workers or the public. Releases could exceed applicable PAG or ERPG levels onsite but not offsite OR Actual or potential major degradation in the level of safety or security that could with further degradation produce a General Emergency</td>
<td>Actual malevolent acts resulting in loss of special nuclear material or a radiological sabotage event that could have impacts beyond the facility boundary but not offsite</td>
<td>Actual or potential major reduction in safety of a shipment. Release may exceed PAG or ERPG levels beyond the exclusion zone* onsite but not at nearest Site boundary</td>
</tr>
<tr>
<td>General Emergency</td>
<td>Actual or imminent catastrophic reduction of facility safety or security systems with potential for the release of large quantities of radiological or nonradiological materials to the environment. Releases reasonably expected to exceed applicable PAG or ERPG levels offsite</td>
<td>Malevolent action resulting in catastrophic degradation of protection systems that could lead to substantial offsite impacts</td>
<td>Actual or imminent catastrophic reduction in safety of a shipment. Release expected to exceed PAG or ERPG levels offsite</td>
</tr>
</tbody>
</table>

*Exclusion zone — the immediate vicinity of the accident

ERPG = Emergency Response Planning Guidelines
PAG = protective action guides
Emergency responses to doses are based on protective action guides (PAGs) that have been adopted by the states of Washington and Oregon and published in EPA-400, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*. These PAGs are intended to apply to projected doses from airborne releases of radioactive materials and subsequent depositions during the early, intermediate, and late phases of an accident. The pathways considered include external gamma and beta dose from direct exposure to airborne materials and from deposited material, and the committed dose to internal organs from inhalation of radioactive material.

The projected dose value for initiating protective actions (evacuation or sheltering) specified by the states of Washington and Oregon is 1 rem TEDE, where the projected dose represents the sum of the effective dose equivalent resulting from exposure to external sources and the committed effective dose equivalent from all significant inhalation pathways during the early phase. The PAG values for committed dose equivalent to the thyroid and skin are 5 and 50 times larger, respectively.

The U.S. Environmental Protection Agency PAGs are stated in terms of committed dose. Dose incurred before initiation of protective actions (and after the early phase of an event) normally are not included when considering whether or not to take protective actions. It is intended that the PAG values be compared to the dose that can be avoided by taking protective actions.

Exposures are based on ERPGs developed and approved by the American Industrial Hygienists Association. The ERPGs are used to determine the appropriate emergency class for exposures to nonradiological releases. The Temporary Emergency Exposure Limit values developed by the Chemical Exposures Working Group of the DOE Subcommittee on Consequence Assessment and Protective Actions are used for chemicals that do not have ERPG values.

Upon declaration of an Alert the Hanford Site ERO will:

- Activate the DOE Hanford EOC and establish communications, consultation, and liaison with offsite agencies.
- Continuously assess pertinent information for RL/ORP decision makers, offsite agencies, the public, and other appropriate entities.
- Conduct appropriate assessments, investigations, or preliminary sampling and monitoring.
- Mitigate the severity of the occurrence or its consequences.
- Prepare for other response actions should the situation become more serious, requiring EROs to mobilize or activate resources.
15 4 3 1 2 2 Site Area Emergency  A Site Area Emergency will be declared when events are predicted, in progress, or have occurred that involve actual or potential major failures of functions needed for protection of workers or the public in accordance with DOE/RL-94-02. A Site Area Emergency will be declared for release of radiological material with a potential dose at the facility boundary greater than or equal to 1 rem TEDE. Site Area Emergency levels for hazardous nonradiological material will be potential exposure to air concentrations greater than or equal to ERPG-2 at the facility boundary but less than ERPG-2 at the Hanford Site boundary (See Tables 15-2 and 15-3).

Upon declaration of a Site Area Emergency, the Hanford Site ERO will perform the same actions as an Alert plus the following:

- Initiation of predetermined protective actions for onsite personnel
- Provision of information to the public and the media
- Implementation of or assistance in any evacuations and sheltering
- Mobilization of appropriate emergency response groups or protective or security forces for immediate dispatch should the situation become more serious

15 4 3 1 2 3 General Emergency  A General Emergency will be declared when events are predicted, in progress, or have occurred that result in the actual or imminent catastrophic reduction of SNF Project facility safety or security systems in accordance with DOE/RL-94-02. A General Emergency will be declared for release of radiological material with a potential dose at the Hanford Site boundary greater than or equal to 1 rem TEDE. General Emergency levels for hazardous nonradiological material will be potential exposure to air concentrations greater than or equal to ERPG-2 at the Hanford Site boundary (See Tables 15-2 and 15-3).

Upon declaration of a General Emergency, the Hanford Site ERO will perform the same actions as a Site Area Emergency plus the following:

- Notification, mobilization, and dispatch of all appropriate emergency response personnel and equipment, including appropriate DOE emergency response assets
- Liaison with offsite agencies for the recommendation of predetermined public protective actions

15 4 3 2 Hanford Transportation Emergency Preparedness Program  Shipments of radiological and nonradiological hazardous material on roadways north of the Wye Barricade are exempt from the Department of Transportation regulations in 49 CFR. Shipments transported south of the Wye Barricade are regulated under 49 CFR unless public access control is extended south of the Wye Barricade (but not beyond the Site boundary) for special case shipments.
The Incident Command System will be used to mitigate transportation incidents that occur on the Hanford Site. Upon notification of the event by POC, the Hanford Fire Department will assume incident command responsibilities. The Emergency Duty Officer (EDO) will have the responsibility for event classification and activation of the Hanford Site ERO.

For transportation incidents involving DOE-owned hazardous materials that occur off the Hanford Site, the POC will provide information to first responders on a 24-hour basis in accordance with 49 CFR requirements. The POC will connect the caller directly with the Transportation On-Call Representative and the EDO, who will provide more detailed information regarding the shipment and follow-on response assistance as appropriate.

**15 4 3 3 Resource Conservation and Recovery Act Emergency** A RCRA emergency is defined as release, fire, or explosion that threatens human health or the environment. The BED/IC, in consultation with an environmental single point-of-contact, will determine whether the incident is a RCRA emergency. The BED/IC will ensure that trained personnel identify the character, source, amount, and extent of the release, fire, or explosion to the extent possible. These activities will be performed with a sense of immediacy and will include all available relevant information.

Hazards posed by the event to human health and the environment will be assessed. The assessments will consider the direct, indirect, immediate, and long-term effects of the incident. Input for the assessments will include sources such as Material Safety Data Sheet toxicity and health information, and results from any personnel monitoring examinations conducted at medical facilities.

If assessment of all available information does not yield a definitive assessment of the danger posed by the incident, a worst-case condition will be presumed and appropriate protective actions and notifications will be initiated. The BED/IC will be responsible to initiate protective actions based on their best judgement of the incident. Environmental notification requirements are provided in Section 15 4 4 2.

**15 4 3 4 Abnormal Event** Events or situations may occur on the Hanford Site that may generate public concern or media interest but not create, or indicate, an emergency condition. RL will work with offsite agencies to maintain criteria that will be used to identify these situations, termed Abnormal Events. Abnormal Event notification requirements are provided in Section 15 4 4 3.

**15 4 3 5 Emergency Action Levels** Using the accident scenarios and consequences identified in the hazards assessments, the observable methods of indicating or recognizing an emergency can be determined using EALs. The EALs are specific, predetermined observable criteria used to detect, recognize, and determine the classification of Hazardous Material Operational Emergencies (see Tables 15-2 and 15-3) identified by the hazards assessments. On determining that an event has occurred or is occurring, the BED/IC will promptly assess the conditions, compare the indications to the facility EALs, and determine the appropriate Hazardous Material Operational Emergency classification.
The level of emergency classification will be used to trigger specific, preplanned, graded responses, such as activation of the Hanford Site ERO and initiation of protective actions and appropriate notifications commensurate with the degree of hazard presented by the event. Actual airborne concentrations are determined by obtaining samples at a specific location using samplers.

15.4.4 Notifications

If an SNF Project-related emergency event were to occur, prompt and accurate emergency notifications would be made to mitigate consequences and to protect the health and safety of workers, the public, and the environment in accordance with Section 5.5.2 of the S/RID (HNF-SD-SNF-RD-001). Notifications will be made in order of urgency with Operational Emergency (Hazardous Material Operational Emergency only) notifications performed first, Environmental notifications performed second, and Abnormal Event (including Base Program Operational Emergency), Unusual Occurrence, and Off-Normal Occurrence notifications performed last.

Procedures will be developed to ensure that notification and reporting requirements are in accordance with DOE O 151.1 and DOE O 232.1A, applicable Federal, state or local requirements, and special agreements with offsite agencies or tribal governments. These procedures will be validated through drills and exercises.

The Unusual and Off-Normal Occurrence categories will be used solely for reporting versus immediate action purposes. Notifications and written reports of incidents meeting occurrence reporting criteria will be made to DOE-HQ and also to offsite entities as requested by RL. Offsite transportation events involving RL/ORP-owned hazardous materials will be reported in accordance with DOE O 151.1 and 49 CFR requirements.

15.4.4.1 Operational Emergency Notification

Prompt and accurate emergency notifications will be made to mitigate consequences and protect the health and safety of workers and the public. The person who witnesses or discovers an abnormal or emergency condition will promptly notify the POC by dialing 911 for fire, medical, and hazardous materials assistance, and the SNF Project facility shift manager/Bed. If the shift manager/Bed is not present, the person will notify the on-call BED. Onsite notifications are required as soon as an event is determined to threaten personnel safety.

The BED/IC ensures activation of appropriate system facility take-cover sirens and direction to the POC to make crash-alarm telephone system notifications. Once the event is classified, the BED/IC will contact the POC and ONC to initiate emergency notifications for onsite and offsite EROs. After being notified of the classification of an emergency event, the ONC duty officer and available staff will make offsite notifications to DOE Headquarters and other potentially affected federal, state, tribal, or local organizations. RL/ORP will oversee the offsite contractor notifications.
The release of public information, which is integrated through the JIC (See Section 15 4 1 3 2), is an integral part of the emergency management program. Offsite agency interfaces and mutual assistance agreements are documented in the memoranda of understanding (DOE/RL-94-02, Appendix B). Information is provided to the public and to employees during the course of an emergency. RL, state governments, and local governments share in this responsibility. Each response organization speaks to its individual area of responsibility.

When operational, the DOE Hanford EOC is responsible for followup notifications when emergencies are reclassified or terminated. If the DOE Hanford EOC is not operational, the followup notifications are made by the IC or SNF Project BED through the POC and ONC.

Procedures will be established and maintained to provide prompt initial notification to workers and emergency response personnel and organizations, including appropriate offsite agencies under the most limiting set of conditions.

Operational Emergencies that also meet RCRA emergency criteria will be categorized in accordance with Section 15 4 3 3 and perform notification in accordance with Section 15 4 4 2.

15 4 4 1 1 Base Program The SNF Project will ensure that the designated point-of-contact (e.g., BED) reports events to the ONC that meet notification criteria based on established procedures. These notifications shall be made as soon as possible (within 30 minutes). The designated point-of-contact, with assistance from ONC personnel, will assess the event information to determine whether the event should be categorized as a Base Program Operational Emergency. If the event meets the Base Program Operational Emergency criteria, the ONC will notify the DOE-HQ EOC within 30 minutes following categorization and the offsite agencies immediately following as part of the Abnormal Event notification (See Section 15 4 4 3).

These notification requirements also apply to offsite transportation events involving DOE-owned hazardous materials.

15 4 4 1 2 Hazardous Material Operational Emergency Notification Hazardous Material Operational Emergency notifications are made to:

- Augment the Site and SNF Project operating staff with personnel in designated response roles to respond to the emergency.
- Activate emergency centers.
- Facilitate public notification by offsite authorities and agencies that have decision-making authority for directing protective actions (e.g., evacuation of local areas).
- Protect site and SNF Project personnel and emergency workers through the provision of information necessary to implement accountability and protective actions such as sheltering, decontamination, and evacuation.

fsar15 sar 15-27 November 1999
15 4 4 1 2 1 Initial Onsite and Offsite Notification  The initial event classification (Alert Site Area Emergency, or General Emergency) will be made by the BED or IC in accordance with established procedures  Immediate notifications will be initiated via the 911 emergency number to request emergency response assistance and to notify onsite personnel via sirens, the onsite crash alarm telephone system, or plant telephone for the purpose of initiating protective actions  The BED/IC will also ensure that a completed copy of the RL Notification Form is transmitted to the ONC in accordance with established procedures  If a facsimile machine is not available, the BED/IC is responsible for ensuring that pertinent information from the RL Notification Form is provided to the ONC

For non-facility events (e.g., onsite transportation incidents, range fires), the EDO is responsible to make the initial event classification, initiating appropriate protective actions, and providing notifications, including information to complete the RL Notification Form, to the ONC  Upon notification from the BED or IC of an emergency event classified Alert, or higher, the ONC will make offsite notifications within 15 minutes to the following

- DOE-HQ EOC
- Benton, Franklin, and Grant Counties, Washington State, and Energy Northwest (WNP-2) via the DOE Crash Alarm Telephone System (hot line)
- Oregon State

The ONC will also initiate the automated Emergency Notification System (ENS) and pager system to activate the DOE Hanford EOC and make onsite notifications, as appropriate to the following

- DOE Hanford senior management on-call
- Emergency Duty Officer (Fluor Daniel Hanford [FDH])
- Pacific Northwest National Laboratory single point-of-contact
- Bechtel Hanford, Inc single point-of-contact
- HEHF single point-of-contact

The ONC will also notify, as applicable to the event, other offsite agencies that may have personnel working in remote locations of the Hanford Site (e.g., personnel at locations without alarm or siren capabilities) within 30 minutes of event declaration  All other notifications will be made as soon as practical  The ONC will maintain a list of agencies to be notified

15 4 4 1 2 2 Reclassification Notification  The BED or IC will reclassify rapidly escalating emergencies until the DOE Hanford EOC is operational  In addition, the BED or IC will provide immediate protective action notification to personnel within their respective geographical area of
responsibility and also provide notification to the POC and ONC via the 911 emergency number regarding the reclassification. The ONC will then notify the offsite EROs of the event reclassification.

When operational, the DOE Hanford EOC will have the responsibility for reclassifying or terminating emergencies, disseminating additional protective action decisions to onsite personnel, and performing offsite notifications that include PARs.

15 4 4 1 2 3 Reports The SNF Project will submit a final report on the emergency to the Occurrence Report Processing System in conjunction with the Final Occurrence Report in accordance with the requirements of DOE O 232 1A following termination of the emergency response.

15 4 4 2 Environmental Notifications Environmental notifications will be made verbally or in writing depending on the type of event. The SNF Project will establish procedures to ensure implementation of environmental notifications in accordance with Federal, state, and local requirements and agreements. The SNF Project will ensure that environmental notification procedures are consistent with the environmental notification process depicted in Figure 5 3 of DOE/RL-94-02.

For incidents involving a spill, release, fire, explosion, or environmental permit exceedence, the SNF Project single point-of-contact will be notified to determine applicability of requirements and perform the necessary environmental notifications including notifying the appropriate Federal, state, and local agencies. The ONC will also be notified for determination if an Abnormal Event notification is also required. The SNF Project will develop any necessary written reports for submittal to RL for review and concurrence.

The FDH environmental single point-of-contact will notify the appropriate Federal, state, and local agencies that the SNF Project facility is in compliance with cleanup activities before operation is resumed.

15 4 4 3 Abnormal Event Notifications The SNF Project will ensure that events meeting the Abnormal Event criteria established by RL are promptly reported to the ONC in accordance with the process in DOE/HFID 232 1B Notification, Reporting and Processing of Operations Information.

15 4 5 Emergency Facilities and Equipment

Emergency facilities provide a location for coordinating emergency response activities. Emergency equipment consists of the materials and tools that may be required to measure, control, or mitigate the consequences of an emergency including equipment required to notify employees of an emergency to facilitate safe evacuation.
15 4 5 1 Spent Nuclear Fuel Project Emergency Facilities  Locations are specified in the SNF Project facility BEPs for establishing an ICP where emergency response personnel may gather to manage onsite emergencies. These locations will be equipped with emergency response materials and equipment that may be used to measure, control, and mitigate the consequences of an emergency.

15 4 5 2 Hanford Site Emergency Facilities  The Hanford Fire Department maintains fully staffed and equipped fire stations in the 100 and 200 East areas. These stations are staffed with personnel and resources for ensuring their ability to respond to a variety of emergencies such as fire, natural disasters, hazardous material releases, and personnel injuries. Hanford Environmental Health Foundation maintains a first aid and medical facility in the 200 East Area. This facility is staffed by trained and qualified medical professionals who are available to provide emergency medical care to injured personnel. Personnel decontamination sites are located in several locations in the 100, 200, 300 and 400 Areas.

The EDF (adjacent to the Kadlec Medical Center, 888 Swift Boulevard, Richland, Washington) contains unique equipment for performing minor medical procedures and decontamination on severely contaminated persons with injuries that are non-life threatening. The locations of emergency facilities are shown in Figure 15-4.

15 4 5 3 U.S. Department of Energy, Richland Operations Office, Emergency Centers Supporting the Spent Nuclear Fuel Project  The DOE Hanford EOC is located in the Federal Building, 825 Jadwin Avenue, Richland, Washington. The DOE Hanford EOC, UDAC, and ONC are dedicated facilities located in the basement of the Federal Building. The JIC is a dedicated facility located on the main floor of the Federal Building in Rooms 157 and 158. Telecommunications equipment, word processing support, and duplicating equipment are provided to support JIC-participating agencies and the media. The JIC also may dedicate the use of the auditorium, portions of the lobby, and other areas in the Federal Building for JIC purposes as needed.

The alternate DOE Hanford EOC would be activated if for any reason the DOE Hanford EOC was not habitable. The alternate DOE Hanford EOC is located in the 2420 Stevens building, rooms 117 and 153.

The POC is located in the 2721E Building in the 200 East Area and acts as a single point of contact for RL/ORP and Hanford Site contractors. The POC monitors the emergency response number (911) and business number (373-3800). The POC provides alarm monitoring, activation of crash alarm telephone systems and sirens, and assisting in dispatch and radio communications for emergency responders. The POC notifies and/or dispatches the following:

- Hanford Fire Department, including ambulance and the Hazardous Material Response Team
- Hanford Patrol
The ONC is a 24-hour operational facility equipped to communicate information regarding occurrences at or affecting the Hanford Site to RL/ORP and Site contractor personnel and to state and local emergency management organizations. The ONC

- Activates the Hanford Site ERO via the automated ENS
- Provides initial notifications via the automated ENS to Grant County residents within the Hanford EPZ
- Provides notifications to the DOE-HQ EOC and state and local emergency management agencies

15 4 5.4 Spent Nuclear Fuel Project Emergency Equipment  Emergency equipment is any equipment or system that may be required as identified by results of the hazards assessments to measure, control, or mitigate the consequences of an emergency or in any way be involved in an emergency response. The SNF Project has fixed emergency systems and equipment and portable emergency equipment, including equipment that is staged or in a designated area for easy retrieval. As required by Section 5.4.2 of the S/RID (HNF-SD-SNF-RD-001), a listing of emergency equipment and locations of emergency equipment are listed in Table 15-4. Detection ranges and types of instruments for radiological and nonradiological hazardous materials will be adequate to cover for emergency conditions as determined in Sections 15 4 2.1 and 15 4 2.2. Hanford Site emergency resources and equipment are listed and described below. Emergency equipment and backup equipment will be inventoried in accordance with approved procedures and inspected in accordance with procedures to ensure that it is available, accessible (away from scene of a potential accident), and operational. The quarterly inventory records will be maintained for one year in accordance with DOE/RL-94-02 requirements.

15 4 5.4.1 Nonradiological Hazardous Emergency Response Equipment  Hazardous material spill kits are used to clean up nonradiological emergency spills.

15 4 5.4.2 Radiological Emergency Response Equipment  Protective clothing and equipment used for response to a radiological emergency is stored in designated storage areas maintained specifically by the SNF Project for emergency events.
### Table 15-4 Emergency Equipment (2 sheets)

<table>
<thead>
<tr>
<th>Type</th>
<th>Capabilities</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Fixed and portable equipment</strong></td>
<td></td>
</tr>
<tr>
<td>Fire control system</td>
<td>Assists in the control of a fire</td>
</tr>
<tr>
<td>- Fire detection and alarm system</td>
<td>Assists in notifying personnel, summoning the Hanford Fire Department, and in fire suppression</td>
</tr>
<tr>
<td>and wet pipe automatic sprinkler</td>
<td></td>
</tr>
<tr>
<td>suppression system</td>
<td></td>
</tr>
<tr>
<td>Safety shower and eyewash station</td>
<td>Assists with personnel decontamination of hazardous (chemical) materials</td>
</tr>
<tr>
<td>Pressure alarms and/or water flow</td>
<td>Assists with notifying personnel of emergency conditions</td>
</tr>
<tr>
<td>alarm</td>
<td></td>
</tr>
<tr>
<td>Evacuation and take cover siren</td>
<td>Assists with notifying personnel of emergency conditions and by the type of siren, expected actions</td>
</tr>
<tr>
<td>Area radiation monitoring system</td>
<td>Alerts personnel to high radiation levels in the K East and/or K West Basin area</td>
</tr>
<tr>
<td>(used in the K Basins only)</td>
<td></td>
</tr>
<tr>
<td>Red crash alarm telephone</td>
<td>Alerts personnel in an emergency and communicates emergency information</td>
</tr>
<tr>
<td>Respiratory protection*</td>
<td>Protects personnel from hazardous chemicals</td>
</tr>
<tr>
<td><strong>Portable emergency equipment</strong></td>
<td></td>
</tr>
<tr>
<td>Fire extinguisher (Types A, B, and C)</td>
<td>Assists in fire suppression</td>
</tr>
<tr>
<td>Portable air compressor (brought in as needed)</td>
<td>Provides backup compressed air to failed compressed air system</td>
</tr>
<tr>
<td>Portable electrical generator (brought in as needed)</td>
<td>Provides backup electrical power where necessary to a facility that has lost normal electrical power</td>
</tr>
<tr>
<td>Hazardous materials spill control kits (unmounted)</td>
<td>Assists with hazardous (chemical) materials stabilization and cleanup following a spill or release</td>
</tr>
<tr>
<td>Command post equipment emergency procedures, checklists (maps and photographs of facilities optional)</td>
<td>Provides area and site-specific emergency information</td>
</tr>
</tbody>
</table>
Table 15-4 Emergency Equipment (2 sheets)

<table>
<thead>
<tr>
<th>Type</th>
<th>Capabilities</th>
</tr>
</thead>
<tbody>
<tr>
<td>Operational event scene equipment</td>
<td>Assists in controlling and mitigating the event</td>
</tr>
<tr>
<td>radiological response vehicle, emergency procedures, duty cards, checklists, maps, photographs of facilities</td>
<td></td>
</tr>
<tr>
<td>Protective clothing and equipment</td>
<td></td>
</tr>
<tr>
<td>Anti-C clothing and personal protective equipment</td>
<td>Provides contamination control (anti-C clothing for radiological and acid gear for any corrosive chemicals)</td>
</tr>
<tr>
<td>Miscellaneous respiratory equipment</td>
<td>Provides respiratory protection, this type of respirator equipment is not considered to be emergency equipment</td>
</tr>
</tbody>
</table>

*Respirators for emergency use will be thoroughly inspected at least once a month and after each use. Records of inspection dates and findings will be maintained.

15 4 5 4 3 Communications Equipment The commercial telephone system is the primary means of communication during an emergency involving the SNF Project. Cellular telephones and radios (i.e., mobile, hand-held, and base stations) will back up the commercial telephone system. Commercial and cellular phones can be used to contact all emergency personnel and organizations for the Hanford Site. If the telephones fail, radio communications to the DOE Hanford EOC can be relayed through the POC.

15 4 5 4 4 Operations Response Emergency Equipment Operations response emergency equipment is stored in designated SNF Project locations. This equipment is used in all types of emergency events and includes ERO staff identification vests, emergency response procedures, personal protective equipment, spill kits, "DO NOT ENTER" tape, yellow rope, radiation rope, flashlights, and first aid kits.

15 4 5 4 5 Fire Fighting Equipment SNF Project facilities use fixed fire alarm systems and, in the majority of hazardous and nonhazardous facilities, a sprinkler system. SNF Projects also have portable fire extinguishers (A-B-C type), located throughout all buildings, that can be used to extinguish small fires. The fire alarm system is used to notify personnel and the fire department of a potential or actual fire situation. The fire sprinkler systems will be used for fire suppression. Other fire fighting equipment is under the control of the Hanford Fire Department.

15 4 5 4 6 Emergency Lighting Emergency lighting is provided for evacuation of SNF Project buildings. The lighting is not intended for reentry into a facility for response activities.
15.4.5.4.7 Operational Alarms  As required by Section 5.4.3 of the S/RID (HNF-SD-SNF-RD-001), an alarm system is required to alert employees of potential emergencies. SNF Project hazardous facilities are equipped with operational alarms for abnormal and emergency conditions, such as the radiation alarms (located in and near radiation areas) and other alarms that provide event indications of a potential or actual emergency. Testing requirements for operational alarms will be contained in controlled procedures.

15.4.5.4.8 Facilities and Site Area Sirens  The 100 K Area evacuation and take-cover sirens are located throughout the 100 K Area and are activated by the 100 K Area BED for the K Basins and Cold Vacuum Drying Facility. The sirens in the 100 K Area may be activated up to 30 minutes during a real event. The sirens are tested monthly.

The 200 East Area also has evacuation and take-cover sirens that will provide coverage for the Canister Storage Building and Interim Storage Area. Various SNF Project buildings are equipped with internal speakers that relay the sirens and public address system announcements.

The SNF Project facility or building manager ensures that preventive maintenance of facility emergency sirens/alarms is performed by the maintenance organization in accordance with procedures as required by DOE/RL-94-02. The FDH emergency preparedness organization will ensure that preventive maintenance is performed on area and river sirens. In order to meet 29 CFR 1910 requirements, the employee alarm system will do the following:

1. Provide warning for necessary emergency action as called for in the emergency action plan, or for reaction time for safe escape of employees from the workplace or the immediate work area, or both.

2. Be capable of being perceived above ambient noise or light levels by all employees in the affected portions of the workplace. Tactile devices may be used to alert those employees who would not otherwise be able to recognize the audible or visual alarm.

3. Be distinctive and recognizable as a signal to evacuate the work area or to perform actions designated under the emergency action plan.

4. The preferred means of reporting emergencies, such as manual pull box alarms, public address systems, radio, or telephones shall be explained to each employee. Emergency telephone numbers shall be posted near telephones, employee notice boards, and other conspicuous locations when telephones serve as a means of reporting emergencies. Where a communication system also serves as the employee alarm system, all emergency messages shall have priority over all non-emergency messages.

5. Procedures for sounding emergency alarms in the workplace shall be established. For those employers with 10 or fewer employees in a particular workplace, direct voice communication is an acceptable procedure for sounding the alarm provided all employees can hear the alarm. Such workplaces need not have a back-up system.
Protective Actions

Protective actions are those actions taken to preclude or reduce the exposure of individuals or the environment impacted by hazards or unsafe conditions during an emergency event at a SNF Project facility as required by Section 5 6 2 of the S/RID (HNF-SD-SNF-RD-001). Protective actions for the SNF Project will reflect the use of chemical ERPGs identified in Section 15 4 3 1. The ERPGs published in the Emergency Response Planning Guidelines (AIHA 1988) are used during an SNF Project facility emergency response to determine protective actions for unique exposures to chemical releases (see Table 15-5). The PAGs also are used during an SNF Project facility emergency response to determine protective actions for unique exposures to radiological releases (see Table 15-2). Published PAGs adopted by the states of Washington and Oregon (EPA-400) will be used in accordance with DOE/RL-94-02.

<table>
<thead>
<tr>
<th>Emergency classification level</th>
<th>ERPG*</th>
<th>Exposure information</th>
</tr>
</thead>
<tbody>
<tr>
<td>Alert</td>
<td>ERPG-1</td>
<td>The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to 1 hour without experiencing other than mild transient adverse health effects or perceiving a clearly defined objectionable odor.</td>
</tr>
<tr>
<td>Site Area Emergency</td>
<td>ERPG-2</td>
<td>The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to 1 hour without experiencing or developing irreversible or other serious health effects or symptoms that could impair their abilities to take protective action.</td>
</tr>
<tr>
<td>General Emergency</td>
<td>ERPG-3</td>
<td>The maximum airborne concentration below which it is believed that nearly all individuals could be exposed for up to 1 hour without experiencing or developing life-threatening health effects.</td>
</tr>
</tbody>
</table>

*For the purposes of applying the emergency class definitions from DOE O 1511 Comprehensive Emergency Management System the terms ERPG and appropriate ERPG exposure levels shall be interpreted to mean a 15 minute time weighted average concentration of the substance in air that equals or exceeds the published ERPG 2 values or its alternative value for that substance.

ERPG = Emergency Response Planning Guidelines

Emergency response procedures include the following types of procedures:

- Activation of internal facility alarms or communication systems
- Area evacuation
The SNF Project is responsible for ensuring that all personnel and visitors within the facility perimeter fences respond appropriately to alarms. Assigned personnel will be provided with specific training on protective actions. The facility ERO will provide area and building sweeps, as necessary to ensure appropriate action is taken.

The Hanford Site emergency management program uses the EPZ concept to focus emergency planning activities. EPZs are designated areas where protective actions may be required. The size of a zone is determined primarily by the expected dispersion distance of a particular concentration of a substance. The two exposure pathways for both radiological and nonradiological hazardous materials are the plume exposure pathway and the ingestion exposure pathway. Table 15-6 presents the plume exposure EPZs for geographical areas on the Hanford Site with potential offsite consequences. The circles in Figure 15-5 illustrate the Hanford Site plume exposure EPZs.

The plume exposure pathway EPZ is the probable area of exposure to a passing cloud, or plume, of the substance potentially resulting in direct contact with the substance through the exterior of the body or through inhalation. The plume exposure pathway EPZ includes the area where emergency planning is conducted (1) to ensure that prompt and effective actions are taken in the event of an emergency (2) to protect onsite personnel, and (3) to ensure public health and safety.
Table 15-6  Hanford Site Area Plume Emergency Planning Zones

<table>
<thead>
<tr>
<th>Location</th>
<th>Type of hazard determining EPZ size</th>
<th>Radius of zone*</th>
</tr>
</thead>
<tbody>
<tr>
<td>100 K Area</td>
<td>Radiological</td>
<td>80 km (50 mi)</td>
</tr>
<tr>
<td>100 N Area</td>
<td>Radiological</td>
<td>50 km (30 mi)</td>
</tr>
<tr>
<td>200 E and W Areas</td>
<td>Radiological</td>
<td>16 km (10 mi)</td>
</tr>
<tr>
<td>300 Area</td>
<td>Radiological</td>
<td>50 km (30 mi)</td>
</tr>
<tr>
<td>400 Area</td>
<td>Radiological</td>
<td>72 km (45 mi)</td>
</tr>
</tbody>
</table>

*For the purposes of EPZ definition the receptor location is defined as the south and/or west shore of the Columbia River.

EPZ = emergency planning zone

The ingestion exposure pathway EPZ is the probable area of exposure to contaminated foodstuffs or water potentially resulting in deposition of the material in various internal organs following ingestion (eating or drinking). The ingestion exposure pathway EPZ for radiological and nonradiological incidents at all Hanford Site facilities corresponds to the 80-km (50-mi) EPZ for Energy Northwest's Nuclear Plant 2. The gray area in Figure 15-6 represents the ingestion exposure EPZ for the Hanford Site.

The protective actions required to minimize the exposure of workers and the public are summarized in the following subsections. Examples of protective actions as a function of accident category and consequences are illustrated in Table 15-7.

Protective actions will be predetermined for onsite personnel and the public and will include the following:

- Methods for controlling, monitoring, and maintaining records of personnel exposures to hazardous materials (radiological and non-radiological)
- Plans for timely sheltering and/or evacuation of workers
- Methods for controlling access to contaminated areas and for decontaminating personnel or equipment exiting the area
- Actions to be taken to increase the effectiveness of protective actions (i.e., heating, ventilation, and air conditioning shutdown during sheltering)
Table 15-7  Example Matrix of Accidents, Accident Types, Consequences, and Protective Actions

<table>
<thead>
<tr>
<th>Accident category</th>
<th>Accident type</th>
<th>Consequence</th>
<th>Protective actions</th>
</tr>
</thead>
<tbody>
<tr>
<td>Radioactive material</td>
<td>Loss of contamination,</td>
<td>Exposure of public workers or</td>
<td>Evacuation sheltering decontamination, relocation, access control, food control,</td>
</tr>
<tr>
<td>release</td>
<td>criticality</td>
<td>environment</td>
<td>personal protective equipment</td>
</tr>
<tr>
<td>Chemical release</td>
<td>Loss of confinement</td>
<td>Exposure of public workers and</td>
<td>Evacuation sheltering decontamination, relocation, access control, food control,</td>
</tr>
<tr>
<td></td>
<td></td>
<td>environment</td>
<td>personal protective equipment</td>
</tr>
<tr>
<td>Natural disaster</td>
<td>Earthquake</td>
<td>Damage to safety features</td>
<td>Evacuation and/or sheltering</td>
</tr>
<tr>
<td>Fire or explosion</td>
<td>Exothermic reactions</td>
<td>Personnel exposure to hazardous</td>
<td>Evacuation decontamination, relocation, access control, personal protective</td>
</tr>
<tr>
<td></td>
<td></td>
<td>materials</td>
<td>equipment (depending on event)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Personnel injuries</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Damage to facility and safety features</td>
<td></td>
</tr>
<tr>
<td>Extrinsic hazard</td>
<td>Airplane crash</td>
<td>Personnel exposure</td>
<td>Evacuation decontamination, relocation, access control, personal protective</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Personnel injuries</td>
<td>equipment</td>
</tr>
<tr>
<td>Transportation</td>
<td>Accident while moving</td>
<td>Direct exposure to ionizing materials</td>
<td>Evacuation decontamination, relocation, access control, personal protective</td>
</tr>
<tr>
<td></td>
<td>radioactive fuel</td>
<td></td>
<td>equipment</td>
</tr>
</tbody>
</table>

- Methods for providing timely PARs, such as sheltering, evacuation, relocation, and food control, to appropriate offsite agencies

- PAGs and ERPGs, prepared in conformance with DOE-approved guidance applicable to the actual or potential release of hazardous materials to the environment, for use in protective action decision making

- The administration of medications

The specific event will determine the initial protective actions at the facility. Other actions may include emergency rescue, decontamination, fire fighting, and medical response. The BED is responsible for event scene protective actions at the SNF Project facility and for initial area protective actions. Immediate offsite protective action decisions (which may include access control, sheltering, and evacuation) and notifications to populations within the plume EPZ are the responsibility of the counties and are primarily provided using the Emergency Alert System. Protective action decisions for the ingestion exposure EPZ are the responsibility of the state, which coordinates implementation of protective actions with the impacted counties.
15 4 6 1 Emergency Evacuation  Planning for emergency evacuations includes determining evacuation routes, transportation needs, and the estimated time when the evacuation sirens would be activated, as well as the possible relocation of emergency facilities in accordance with DOE/RL-94-02. SNF Project facility preparations include establishing safe transportation and notifying personnel as to what protective actions they need to perform and what actions are required when evacuation sirens are activated.

The emergency evacuation process is described in the individual BEPs. Evacuation procedures include activating evacuation sirens, communicating evacuation routing, transporting personnel, and initiating protective actions for personnel immediately adjacent to the event. All personnel are trained in these evacuation procedures. The SNF Project facilities are also equipped with interior evacuation sirens and crash alarm telephones. Facility evacuation routes are designated by the shift manager (BED) according to the event and meteorological conditions. Evacuation routes for the Hanford Site are specified in Figure 7-3 of DOE/RL-94-02. Once determined, the evacuation routes are communicated to other Hanford Site facilities.

As required by DOE/RL-94-02, personnel accountability is conducted following implementation of protective actions (Section 15 4 6) to assure that employees are properly accounted for and the affected facility has been evacuated as necessary. Accountability will be conducted at the staging area to ensure all employees are properly accounted for.

15 4 6 2 Sheltering (Take Cover)  Depending on factors such as the type of materials released and the facility-specific conditions, sheltering (take cover) may be the preferred protective action when it will provide protection equal to or greater than evacuation.

15 4 6 3 Relocation of Emergency Facilities  Unsafe conditions onsite may cause emergency facilities to relocate to alternate locations.

15 4 6 4 Access Control and Traffic Control  A single point of entry and egress is established to control access to the event scene and the incident control personnel. Roadblocks established in safe locations prevent entry into the event scene and downwind locations.

15 4 6 5 Use of Protective Equipment  Protective clothing and respiratory equipment may be used to provide a protective barrier to minimize exposure to loose surface and airborne contamination during response to emergencies.

15 4 6 6 Food Control  Intervention levels for offsite foodstuffs are based on Food and Drug Administration guidelines as described in the Washington State procedure, Derived Intervention Levels (WASH 1991). The DOE Hanford EOC Policy Team provides recommendations to the offsite agencies as applicable.

15 4 6 7 Personnel Monitoring  Personnel are monitored for radiological contamination at the scene to determine appropriate actions (i.e., segregate them from uncontaminated personnel, secure areas where contamination has been spread, transport contaminated personnel to decontamination facility).
15 4 6 8 Exposure and Contamination Controls  Exposure and contamination controls prevent overexposure of response personnel during the initial response phases and thereafter and reduce or prevent the spread of contamination by responders and/or vehicles.

15 4 6 9 Event Scene Controls  Event scene controls and response actions are established at and controlled from the field operations control area.

15 4 6 10 State Highway 240 Closure  The Washington State Patrol has developed preplanned protective actions to implement closure of State Highway 240 depending on the event classification. If the Washington State Patrol does not have sufficient resources to accomplish this task, the patrol can request assistance to set up road blocks from the Benton County Sheriff’s Office and Hanford Patrol.

15 4 6 11 Emergency Rescue  The Hanford Fire Department, including paramedics, provides emergency rescue services and assists injured or severely ill personnel from work locations in the SNF Project.

15 4 6 12 Decontamination  Uninjured, contaminated personnel are decontaminated in designated Hanford Site decontamination facilities. Patients with injuries that are not life threatening but who are severely contaminated will be cared for in the EDF, which provides for both isolation and decontamination. Severely injured personnel will be taken directly to a local hospital emergency room.

15 4 6 13 Medical Support  The Hanford Site medical contractor provides professional medical help for the entire Hanford Site. Doctors and nurses are available for emergency assistance at all times. The medical staff is trained to work with personnel who have been contaminated by a radioactive source or exposed to a hazardous material. Hospital services are available at Kadlec Medical Center in Richland with backup service at Kennewick General Hospital in Kennewick and Our Lady of Lourdes Health Center in Pasco.

15 4 7 Training and Exercises

Requirements in Section 5.2 of the S/RID (HNF-SD-SNF-RD-001), DOE/RL-94-02, and 29 CFR 1910 require that the emergency organizations are formed, trained, and tested to ensure recognition and classification of emergencies and the implementation of protective actions. The SNF Project Emergency Response Training Program ensures the readiness of the SNF Project emergency preparedness organizations as well as general employee awareness of hazards and response actions (see Table 15-8).
**Table 15-8 Emergency Preparedness Training Courses**

<table>
<thead>
<tr>
<th>Training course</th>
<th>Frequency</th>
<th>Course summary</th>
</tr>
</thead>
<tbody>
<tr>
<td>General Employee Training</td>
<td>Annually</td>
<td>Training provides basic emergency preparedness response procedures to all SNF Project employees</td>
</tr>
<tr>
<td>Facility ERO Training</td>
<td>Prior to assignment, annually thereafter</td>
<td>Training addresses emergency procedures, responsibilities, and command and control for members of the SNF Project Facility ERO (i.e., BED and support staff)</td>
</tr>
<tr>
<td>Incident Command Organization Training</td>
<td>Prior to assignment, annually thereafter</td>
<td>Training addresses the roles, responsibilities, and authorities for the respective position within the Incident Command Organization</td>
</tr>
<tr>
<td>Emergency Operations Center training</td>
<td>Prior to assignment, annually thereafter</td>
<td>Training provides an overview of the ERO and specific emergency center operation for SNF Project staff assigned to the DOE Hanford EOC</td>
</tr>
<tr>
<td>Visitor, Vendor, Subcontractor, Consultant, and Regulatory Agency Personnel Training</td>
<td>Prior to badging</td>
<td>Training provides safety, security, and emergency preparedness information to visitors, vendors, subcontractors, consultants, and regulatory agency personnel</td>
</tr>
</tbody>
</table>

*BED = building emergency director*

*DOE = U.S. Department of Energy*

*EOC = Emergency Operations Center*

*ERO = Emergency Response Organization*

**General Employee training** is provided to employees who may have to take protective actions in the event of an emergency when they are employed when their expected actions change, or when the SNF Project facility emergency plan or response procedures change. This training will also include participation in drills and exercises. Refresher training will be provided on an annual basis to employees likely to witness a hazardous material release and who are required to notify authorities of the event. Site personnel will be provided information on facility-specific emergency response documentation including emergency signals, basic instruction, and emergency response structure.

All visitors, vendors, contractors, and consultants will receive safety and emergency preparedness training as part of visitor orientation to the Hanford Site. This orientation is required before badging and is offered on a drop-in basis. As appropriate, RL offers training.
programs to offsite organizations that perform emergency tasks. Training may include facility-specific orientations, hazards information, and review of emergency response procedures.

15 4 7 1 Spent Nuclear Fuel Project Emergency Response Training Personnel in assigned roles of the Incident Command Organization (Table 15-1) will attend Incident Command and task-specific training on an approved equivalency prior to assignment and annually thereafter. Refresher training will be provided annually to these SNF Project employees. This will include lessons learned from past drills and exercises, changes to plans and procedures, and lessons learned from emergencies at DOE and other industrial facilities.

The training will include:

- Duties contained within DOE-0223, Hanford Emergency Plan Implementing Procedures and supporting facility-specific emergency response procedures.
- An overview of the Hanford Incident Command System including roles and responsibilities.
- An overview of the facility hazards and hazard control measures specified in accident scenarios in authorization basis documentation and supporting emergency response procedures.

The SNF Project Emergency Response Training Program is based on DOE-0223, the SNF Project facility BEPs, and emergency response procedures. SNF Project managers are responsible for ensuring that all of their personnel receive incident command and task-specific training or approved equivalency prior to being assigned to the Incident Command Organization and annually thereafter. Training will include roles, responsibilities, and authorities for the respective position within the Incident Command Organization. Personnel supporting the emergency response at the event scene will have first aid training (including cardiopulmonary resuscitation and bloodborne pathogen training) and self-contained breathing apparatus training.

Personnel directing or supervising response actions will be trained for all tasks they assign to be performed and have the same level of qualification for emergency response as the personnel being directed. Additional training requirements are provided below. Training requirements for administrative facilities, non-hazardous facilities, and hazardous facilities are contained in the respective subsections of Section 12.0 of DOE/RL-94-02.

15 4 7 1 1 Building Emergency Directors Training The BEDs, and alternates, will attend approved emergency response training prior to assignment to an ERO. The BED, or alternates, provides an orientation to the staging area managers. The emergency preparedness training is a mix of classroom instruction, tabletop exercises or walkthroughs, and drills. The training programs are systematic and performance based and developed using performance objectives that place emphasis on team training and facility-specific emergency response scenarios based on the hazards assessments, past events, and the lessons learned from those events, past drills and exercises, and changes to plans and procedures. Training for BEDs includes the shift.

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manager/operational engineer certification program and SNF Project Emergency Response Training

154712 Staging Area Managers and Personnel Accountability Aides Training

Staging area managers and personnel accountability aides for administrative, nonhazardous, and hazardous facilities receive the SNF Project nonhazardous ERO training prior to assignment and annually thereafter.

15471.3 Other Emergency Response Support Personnel Training

Skilled Support Personnel and Specialist personnel who are not permanent members of the ERO are not required to meet the above ERO training requirements. However, other training is required to ensure that such personnel are protected against hazards that may be present at the event scene.

15471.3.1 Skilled Support Personnel

Personnel who are needed temporarily to perform immediate emergency support will be given an initial briefing prior to their participation in any emergency response. This initial briefing will include instruction in the wearing of appropriate personal protective equipment, what hazards are involved, and what duties are to be performed. Any other safety and health precautions will be used to assure the safety and health of these personnel.

15471.3.2 Specialist Personnel

Personnel who in the course of their regular job duties are trained in the hazards of specific hazardous (radiological and nonradiological) substances and who may be called upon by the BED/IC to provide technical advise or assistance shall receive training or demonstrate competency in the area of their specialization annually.

15472 Spent Nuclear Fuel Project Emergency Drill Program

Emergency drills are conducted to provide training for the emergency staff in the use of emergency facilities equipment, procedures, and communication channels for credible emergencies. These drills are developed and conducted by the SNF Project emergency preparedness personnel. SNF Project administrative procedures provide the process for this training.

Crew drills will be performed as scheduled by the appropriate SNF Project operations organization and the SNF Project emergency planning organization. These drills are performed on-shift by operations personnel for response skill training, evaluation, and improvement purposes. Systems will be in place to track corrective actions as a result of drills. SNF Project administrative procedures provide the process for this training.

Emergency preparedness drills will use the appropriate Emergency Plan Implementing Procedures (DOE-0223) to demonstrate the following:

- Implementation and coordination of facility and/or area (e.g., 100 Area, 200 Area, 300 Area) and protective actions such as take cover or evacuation.
- Event recognition and classification.
Event mitigation

Emergency and environmental notifications and communications

Interface with other Incident Command Organization functions and other affected facilities

Senior SNF Project management will be provided with a predrill and post-drill report for all emergency drills and for corrective actions status. DOE/RL-94-02 requires that operations and/or emergency preparedness drills be conducted with a frequency sufficient to provide proficiency and complete confidence in response capability.

15 4 7 3 Hanford Emergency Exercise Program The Hanford Emergency Exercise Program organization administers the site emergency exercise program. The Hanford Emergency Exercise Program organization develops and maintains a five-year schedule to ensure that each hazardous facility participates in one exercise with the EOC. Offsite agencies, including DOE Headquarters, and appropriate federal, state, tribal, and local organizations, are invited to participate in the annual DOE-RL field exercise and preparatory tabletop exercises to improve integration and coordination of the numerous emergency programs. Systems are in place to track corrective actions as a result of emergency exercises. Additional specifics of the exercise program is provided in Section 13.0 of DOE/RL-94-02.

15 4 7 4 Training Program Evaluation The effectiveness of the training program is evaluated by written examinations, qualification cards (knowledge-performance objectives), and emergency drills and exercises. Drills and exercises for this program will be of sufficient scope and detail to ensure the demonstration of adequate response capability. Drills will emphasize facility-specific emergency events and response activities will minimize generic nonspecific simulations. Such drills will be conducted in accordance with approved procedures. The effectiveness of the emergency preparedness training program will be evaluated during the conduct of drills and exercises.

15 4 8 Reentry and Recovery

Provisions for emergency event termination, facility reentry, transition from an emergency organization to a recovery organization, and recovery provisions are summarized in the following subsections.

15 4 8 1 Emergency Event Termination For an event in which the DOE Hanford EOC has not been activated, an emergency can officially be terminated after applicable criterion has been met, and the BED and the IC agree that termination can be declared. However, in an event where the DOE Hanford EOC has been activated, termination occurs after all applicable criteria have been met, concurrence between the SNF Project staff and RL/ORP has been obtained, and the BED, IC, and SED agree on termination and information is communicated to the RL/ORP Emergency Manager. The RL/ORP Emergency Manager will then coordinate the termination.
recommendation with the state and county representatives and make the official emergency termination declaration

15 4 8 2 Facility Reentry Provisions  Authorization for reentry at any time during the emergency, including cases where immediate action is required for saving a life, minimizing injury, or minimizing damage, is the responsibility of the BED and IC. Authorization for reentry during the recovery phase is the responsibility of the onsite recovery manager.

15 4 8 3 Transition from Emergency Organization to Recovery Organization  A recovery plan will be developed when necessary. A recovery plan will be needed following an event when further risk could be introduced to personnel, the facility, or the environment through recovery action and/or to maximize the preservation of evidence. Depending on the magnitude of the event and the effort required to recover from it, recovery planning may involve personnel from RL/ORP and other contractors. If a recovery plan is required, it will be reviewed by appropriate personnel and approved by the manager of recovery operations. Restart of operations will be performed in accordance with the approved plan. Onsite and offsite emergency organizations will develop recovery staffing plans necessary to return the affected facility and surrounding areas to normal after termination of the event.

When the emergency phase is complete, a special recovery organization may be appointed at the discretion of RL/ORP, in accordance with Section 9.2 of DOE/RL-94-02, to restore conditions to normal. The makeup of this organization depends on the extent of the damage and its effects. The onsite recovery organization is appointed by the SNF Project representative to the SMT.

RL/ORP will determine the type and extent of accident investigations to meet the requirements of DOE O 225 1A, Accident Investigations. In addition, the SNF Project will assist RL/ORP in performing an investigation of the emergency root causes and corrective actions in accordance with DOE requirements to prevent recurrence.

15 4 8 4 Recovery Provisions  The RL/ORP directs the recovery planning for Hanford Site facilities and supports the offsite recovery efforts of federal, state, and local agencies. The recovery support team consists of sufficient staff, including facility staff to perform functions applicable to the situation.

15 5 DOCUMENT CONTROL

As required by DOE/RL-94-02, the BEP and implementing procedures are controlled-distribution documents. The document control system will ensure that controlled copies are up to date and available at locations where needed in an emergency. The implementing procedures will be reviewed annually and revised as necessary.
Emergency procedures will provide for documentation of emergency records that contain information for review and reconstruction of major communications and activities taken during an emergency, including logs and documentation produced by the DOE Hanford ERO

15.6 REFERENCES


49 CFR, Title 49, Code of Federal Regulations

AIHA, 1988, Emergency Response Planning Guidelines, American Industrial Hygienists Association, Akron, Ohio

Atomic Energy Act of 1954, 42 U.S.C 2011, et seq


DOE Order 5400 1, General Environmental Protection Program U.S. Department of Energy, Washington, D.C.


WASH, 1991 Derived Intervention Levels, Olympia Washington
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Figure 15-1  Linking the Spent Nuclear Fuel Project Safety Analysis Report to the Emergency Preparedness Program

SNF Project facility consequences
(Chapter 3 0 Hazard and Accident Analyses)

SNF Project Emergency Preparedness Program

SNF Project Facility Building Emergency Plan

RU/ORP Emergency Preparedness Program

State and Counties Emergency Preparedness

DOE HQ Emergency Preparedness Program

Tribes Other State and Federal Agencies

DOE HQ = U.S. Department of Energy Headquarters
RU/ORP = U.S. Department of Energy Richland Operations Office/Office of River Protection
SNF = spent nuclear fuel

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Figure 15-2 Incident Command System at Hanford
Figure 15-3  Hanford Site Emergency Communication Chain During Declared Emergencies
Figure 15-4  Hanford Site Emergency Centers and Fire Stations
Figure 15-5  Plume Exposure Emergency Planning Zones
CHAPTER 16 0

PROVISIONS FOR DECONTAMINATION AND DECOMMISSIONING
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## 16.0 PROVISIONS FOR DECONTAMINATION AND DECOMMISSIONING

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<th>Description</th>
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<td>ALARA</td>
<td>as low as reasonably achievable</td>
</tr>
<tr>
<td>CERCLA</td>
<td><em>Comprehensive Environmental Response Compensation and Liability Act (CERCLA) of 1980</em></td>
</tr>
<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
</tr>
<tr>
<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
</tr>
<tr>
<td>D&amp;D</td>
<td>decontamination and decommissioning</td>
</tr>
<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>NEPA</td>
<td><em>National Environmental Policy Act (NEPA) of 1969</em></td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
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</table>
16 0 PROVISIONS FOR DECONTAMINATION AND DECOMMISSIONING

16 1 INTRODUCTION

This chapter describes provisions that facilitate decontamination and decommissioning (D&D) of the Spent Nuclear Fuel (SNF) Project facilities after completion of the fuel retrieval, processing, and storage mission. These provisions include guidance on the description of the conceptual D&D plan for the SNF Project facilities. This chapter includes the following:

- Description of design features incorporated in a facility to facilitate future D&D of the facility
- Description of operational considerations to facilitate future D&D
- Description of the conceptual D&D plan

The Cold Vacuum Drying Facility (CVDF) and Canister Storage Building (CSB) and the major modifications to the K Basins and their associated systems and operations have been designed to minimize the spread of contamination and to facilitate future D&D activities. These design features and operational considerations consist of multiple-confinement layers, compartmentalized process areas and systems, modular equipment, and between-processing decontamination that are intended to minimize final building and equipment decontamination and environmental restoration activities.

The process to develop D&D plans for the SNF Project facilities is discussed in Section 16 3.

16 2 REQUIREMENTS

Specific requirements applicable to this chapter include:

- *Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) of 1980, as amended*
- *National Environmental Policy Act (NEPA) of 1969*
- *DOE Order 5820 2A, Radioactive Waste Management, Chapter V "Decommissioning of Radioactively Contaminated Facilities"*
The following are applicable to the safety basis for the CVDF and CSB only

- **Title 10, Code of Federal Regulations, Part 835, "Occupational Radiation Protection" (10 CFR 835), Section 835 1002(d)**

16.3 DESCRIPTION OF CONCEPTUAL PLANS

This section describes the design features and operational considerations pertinent to SNF Project facilities that will facilitate decontamination and ultimate decommissioning and environmental restoration activities. Design and operating practices that minimize the production of contamination and confine it to designated areas are important for final D&D considerations. Additionally, this section describes the conceptual D&D plan for the SNF Project facilities.

16.3.1 Design Features

SNF Project facilities include design features to facilitate D&D. These design features control the location of the planned contamination and minimize the potential for spread of contamination. Planning and designing for reducing the volume of contaminated areas and precluding inaccessible areas where contamination may reside allows for D&D operations to be conducted with less residual risks and in a more cost effective manner. Typical design features that facilitate D&D include the following:

- Use of modular, separable confinements for radioactive and other hazardous materials to preclude contamination of fixed portions of the structure.

- Use of localized liquid transfer systems that avoid long runs of buried contaminated piping.

- Location of exhaust filtration components of the ventilation systems at or near individual enclosures to minimize long runs of internally contaminated ductwork.


• Equipment that precludes, to the extent practicable, the accumulations of radioactive or other hazardous materials in relatively inaccessible areas including curves and turns in piping and ductwork

• Use of materials that reduce the amount of radioactive and other hazardous materials requiring disposal and that are easily decontaminated

• Designs that ease cut-up, dismantlement, removal, and packaging of contaminated equipment from the facility

• Use of modular radiation shielding

• Use of lifting lugs on large tanks and equipment

• Fully drainable piping systems that carry contaminated or potentially contaminated liquids

• Use of metal liners, as required

• Use of smooth finishes (absence of cracks, crevices, and joints to the extent possible) to prevent contaminated material accumulation in inaccessible areas

• Finishes that enhance decontamination of surfaces

Facility-specific design features are described in the facility annexes to the SNF Project Final Safety Analysis Report (FSAR)

16 3 2 Operational Considerations

During normal operating conditions, operating practices will minimize the production of radiological contamination and confine it to designated areas. During the potential accident scenarios identified in Chapters 3 of the facility FSAR Annexes, controls are in place as described in Chapters 4 of the facility FSAR Annexes that will prevent or mitigate the spread of contamination. The likelihood of uncontrolled contamination being present in the SNF Project facilities or contamination being released to the environment during SNF retrieval, processing, and storage is small, thereby aiding eventual D&D cleanup activities. Operational considerations that facilitate D&D include the following:

• Implementation of the radiation protection program which includes elements such as the as low as reasonably achievable (ALARA) policy and program, radiological monitoring, contamination control, access control, and work planning (described in Chapter 7) to minimize the spread or release of contamination.
Implementation of the radioactive and hazardous management program which includes elements such as solid, liquid, and airborne waste handling, treatment, and disposal, and waste minimization practices that reduce the generation of wastes and minimize the spread or release of contamination (described in Chapter 9).

Facility-specific operational considerations are addressed in the facility FSAR Annexes.

16.3.3 Decommissioning

The SNF Project facility shutdown, deactivation, and eventual decommissioning process described in this section provides conceptual D&D plans for the SNF Project facilities. The SNF Project facilities may be identified for reuse instead of a direct path that culminates in facility decommissioning following the end of their present missions. This case is not discussed in this chapter because reuse would require a similar process like that required for the construction, retrofit, and operations of the SNF Project facilities for their current mission of SNF retrieval, processing, and interim storage of SNF.

When operations cease at the SNF Project facilities, the follow-on activities, whether they are associated with shutdown, deactivation, or decommissioning, will be different than those described and evaluated for normal operations in the facility FSAR Annexes. An assessment of these activities based on an evaluation of the type and magnitude of hazards and complexity of the operations conducted in a facility during a specific life-cycle phase is required as described in documentation such as the shutdown, deactivation, surveillance and maintenance, and decommissioning plans identified below.

When a facility's mission has been completed and there are no other identified missions or programs to support, a facility can be shutdown or placed in a standby condition. A facility status change requires U.S. Department of Energy (DOE), Richland Operations Office concurrence. Several options with varying degrees of plant and equipment layup are described in HNF-PRO-1794, Facility Shutdown, Standby, and Transfer. This procedure provides guidance for the preparation of a facility shutdown or standby plan which provides the overall technical guidance for those activities associated with the shutdown or standby. The plan will address the retention of documentation, continued operations of appropriate safety and fire protection equipment, conduct of radiation surveys, reassignment or disposition of property and other activities associated with facility shutdown. A request to transfer to the surplus facility program may take place at this stage.

A facility will undergo deactivation and prepare a deactivation plan once the facility has been formally declared a surplus facility. The deactivation plan is described in HNF-PRO-443, Facility Deactivation Requirements, and provides guidance for the goals, activities, and organizational responsibilities associated with the transition of the surplus facility to a deactivated status. Work includes the removal of all radioactive and hazardous material to the extent possible and the identification and documentation of the material that cannot be removed at this stage. Contamination is also removed to the extent possible or is identified and appropriate confinement.
controls are established. A turnover package is developed that documents the configuration for transfer to D&D. This package includes all of the necessary approved NEPA documentation, safety analysis documents, and configuration documentation.

The SNF Project is not responsible for decommissioning activities, but decommissioning will take place under the DOE Office of Environmental Management program as implemented by the designated onsite contractor. The general stages of a decommissioning project, as identified in DOE/EM-0246, *U.S. Department of Energy Office of Environmental Management Decommissioning Resource Manual*, include the following:

- **Predecision** – Work includes deactivating the facility and establishing a surveillance and maintenance program for care of the facility until actual decommissioning can be performed. After facility deactivation, a preliminary characterization to obtain the general nature and extent of contamination and a preliminary hazard analysis to determine the specific hazards remaining in the facility are performed to determine the appropriate features of the surveillance and maintenance program.

- **Determination of action** – When the decision to proceed with decommissioning is made and the appropriateness of CERCLA reviewed, the scope of the project is defined and the initial cost, schedule, and technical baselines are established in the decommissioning project plan.

- **Choosing the decommissioning alternative** – Additional information is collected and analysis performed to identify the decommissioning alternatives that are addressed in either the CERCLA or the NEPA process. A preferred decommissioning alternative is selected and documented through this process.

- **Engineering and planning** – Engineering and planning work is performed to prepare a decommissioning plan that can be considered Title II, Detailed Engineering, for the project. The decommissioning plan replaces the decommissioning project plan. A readiness review is performed.

- **Performance of decommissioning operations** – Field work is carried out to achieve the end criteria as stated in the decommissioning plan and final radiological, and chemical surveys are conducted to demonstrate meeting the criteria.

- **Post-decommissioning action** – Follow-up actions may include ongoing remedial action programs. Or, as with the case of the entombment option, long-term surveillance and maintenance may be appropriate.

All efforts to keep contamination contained and localized and to prevent its spread to additional items of equipment and other locations will help to minimize the job of decommissioning. Much of the support equipment, such as air compressors and water supplies, and much of the heating, ventilation, and air conditioning equipment, should remain radiologically clean and be relatively simple to dispose of using routine demolition techniques. After completion...
of fuel removal and cold vacuum drying operations, it is anticipated that the K Basins and CVDF will be shutdown, deactivated, and placed in a surveillance and maintenance mode before final dispositioning. This would include removal of all contaminated liquid effluents and flushing and cleaning of liquid effluent systems (e.g., tanks, piping). All loose contamination would be removed from contaminated air systems (e.g., ducting, filters, filter housings). Loose contamination on structure surfaces (e.g., floors, walls, ceilings) would be decontaminated such that all that remains is fixed contamination. Facility decontamination waste will be disposed of at the appropriate waste disposal facilities.

Administrative areas, such as control rooms, kitchens, change facilities, and other structures and equipment located outside of the contaminated areas should remain radiologically clean and be relatively simple to dispose of using routine demolition techniques. Decommissioning of the remaining facility would involve decontamination of the concrete and structure if required, and/or disposal as low-level waste.

All decommissioning plans will comply with DOE Order 5820 2A, Chapter V, "Decommissioning of Radioactively Contaminated Facilities," DOE/EM-0246, or the latest DOE requirements and guidance in effect. The plan will address the restoration of the area and the reduction of any contamination to the level applicable to the preferred decommissioning alternative.

Operational records (e.g., design drawings and modifications, characterization data on contamination levels, prior decontamination activities, and incident reports required by DOE orders) for the SNF Project facilities and their related process equipment are maintained by SNF Project Operations for use in preparing decommissioning plans.

Final decommissioning plans will include detailed characterization and treatment of waste quantities generated during D&D activities. Based on anticipated contamination levels and cleansing techniques, the risk implication for decontamination is low.

During D&D activities, there will likely be some processing of hazardous or radioactive materials generated by cleanup of the SNF Project facilities. Low-level radioactive waste will probably be shipped to onsite disposal. Transuranic wastes will be packaged and certified for permanent disposal in an approved facility. Hazardous waste will be disposed of at a permitted hazardous waste treatment, storage, and disposal facility. The specific treatment, storage, and disposal facility has not been identified.

16.4 REFERENCES


HNF-PRO-443, Facility Deactivation Requirements, Rev 0, Fluor Daniel Hanford, Incorporated, Richland, Washington

HNF-PRO-1794, Facility Shutdown Standby and Transfer, Rev 0, Fluor Daniel Hanford, Incorporated, Richland, Washington


CHAPTER 17 0

MANAGEMENT, ORGANIZATION, AND INSTITUTIONAL SAFETY PROVISIONS
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<td>as low as reasonably achievable</td>
</tr>
<tr>
<td>AMS</td>
<td>Office of Assistant Manager for Engineering and Standards</td>
</tr>
<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
</tr>
<tr>
<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
</tr>
<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
</tr>
<tr>
<td>DOE-HQ</td>
<td>U.S. Department of Energy, Headquarters</td>
</tr>
<tr>
<td>DOE-RL</td>
<td>U.S. Department of Energy, Richland Operations Office</td>
</tr>
<tr>
<td>ES&amp;H</td>
<td>Environment, safety, and health</td>
</tr>
<tr>
<td>FDH</td>
<td>Fluor Daniel Hanford, Incorporated</td>
</tr>
<tr>
<td>FSAR</td>
<td>Final Safety Analysis Report</td>
</tr>
<tr>
<td>IRP</td>
<td>Independent Review Panel</td>
</tr>
<tr>
<td>ISA</td>
<td>Interm Storage Area</td>
</tr>
<tr>
<td>ISMP</td>
<td>Integrated Safety Management Plan</td>
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<tr>
<td>ISMS</td>
<td>Integrated Environment, Safety, and Health Management System</td>
</tr>
<tr>
<td>NEPA</td>
<td><em>National Environmental Policy Act of 1969</em></td>
</tr>
<tr>
<td>NRC</td>
<td>U.S. Nuclear Regulatory Commission</td>
</tr>
<tr>
<td>ORR</td>
<td>Operational Readiness Review</td>
</tr>
<tr>
<td>QAPP</td>
<td>Quality Assurance Program Plan</td>
</tr>
<tr>
<td>PHMC</td>
<td>Project Hanford Management Contract</td>
</tr>
<tr>
<td>RRT</td>
<td>Regulatory Requirements Team</td>
</tr>
<tr>
<td>SAR</td>
<td>Safety Analysis Report</td>
</tr>
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<td>SFO</td>
<td>U.S. Department of Energy Office of Spent Nuclear Fuels</td>
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<td>SNF</td>
<td>Spent Nuclear Fuel</td>
</tr>
<tr>
<td>SRB</td>
<td>Safety Review Board</td>
</tr>
<tr>
<td>S/RID</td>
<td>Standards/Requirements Identification Document</td>
</tr>
<tr>
<td>SSC</td>
<td>Structure, System, and Component</td>
</tr>
<tr>
<td>TSR</td>
<td>Technical Safety Requirement</td>
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<tr>
<td>USQ</td>
<td>Unreviewed Safety Question</td>
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<td>VPP</td>
<td>Voluntary Protection Program</td>
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17 0 MANAGEMENT, ORGANIZATION, AND INSTITUTIONAL SAFETY PROVISIONS

17 1 INTRODUCTION

The Spent Nuclear Fuel (SNF) Project was formed to address the need to move SNF from the present degraded storage conditions in the 100 K Area K Basins to the Cold Vacuum Drying Facility (CVDF) for conditioning and to the Canister Storage Building (CSB) for safe interim storage until final disposition of the SNF is decided at the national level. In addition, the SNF Project has responsibility for stewardship of SNF from locations on the Hanford Site other than the K Basins. SNF from other locations will be stored at the Interim Storage Area (ISA) adjacent to the CSB.

The organizational structure, responsibilities, and interfaces that support safe design, construction, and operational activities of the SNF Project facilities are identified in this chapter. In addition, safety management policies and programs are discussed in sufficient detail to demonstrate the safety conscious environment of the SNF Project. In those cases where policies, programs, and practices important to safe operation are described in detail in other Hanford Site documents, the information is summarized in this chapter and the documents are referenced. The detailed programs and procedures described in referenced documents may be changed without further U.S. Department of Energy (DOE) approval to the extent that the changes do not constitute an unreviewed safety question (USQ) as defined in DOE Order 5480 21, Unreviewed Safety Questions.

17 2 REQUIREMENTS

The requirements that form the basis for management, organization, and institutional safety are found in HNF-SD-SNF-RD-001, Spent Nuclear Fuel Project Standards/Requirements Identification Document. The specific requirements applicable to this chapter include:

- DOE O 232 1A, Occurrence Reporting and Processing of Operations Information
- DOE O 425 1, Startup and Restart of Nuclear Facilities
- DOE Order 5480 19 Conduct of Operations Requirements for DOE Facilities
- DOE Order 5480 20A, Personnel Selection Qualification and Training Requirements for DOE Nuclear Facilities
- DOE Order 5480 21, Unreviewed Safety Questions
17.3 ORGANIZATIONAL STRUCTURE, RESPONSIBILITIES, AND INTERFACES

In accordance with DOE Order 5480 23, this section presents the organizational structure of the SNF Project responsible for design, construction, and operation. This includes organizational responsibilities and interfaces between internal and external organizations. The responsibilities of interfaces are described to a level of detail sufficient to identify the coordination and communication of safety issue discovery, management, coordination, and resolution.

The SNF Project supports the mission to clean up the Hanford Site by managing and reducing hazards associated with the Hanford Site SNF inventory. The SNF Project facilities consist of the K Basins, CVDF, CSB, and ISA. Design and construction of K Basins modifications and the CVDF will allow commencement of SNF removal from the K Basins and provide cold vacuum drying at the CVDF followed by storage of the dried SNF in the CSB while awaiting final disposition in a federal repository. The CSB will provide safe, economical, and environmentally sound storage for the SNF.

17.3.1 Organizational Structure

Fluor Daniel Hanford, Incorporated (FDH) is responsible to DOE, Richland Operations Office (DOE-RL), for planning, integrating, and managing SNF Project activities, including programs, projects, and operations. FDH is supported by subcontractors collectively referred to as the Project Hanford Management Contract (PHMC) team, and other companies referred to as enterprise companies. In addition, other onsite contractors, (e.g., Pacific Northwest National Laboratory, Hanford Site medical contractor), have existing contracts with DOE-RL for Hanford Site support services. Figure 17-1 depicts the FDH organization, the roles of the DOE-RL, FDH PHMC Team, and enterprise companies are shown in Figure 17-2. As shown in Figures 17-1 and 17-2, the PHMC management and integration roles are divided between FDH, major subcontractors, and the enterprise companies.
As noted in Figure 17-1, the FDH Environment, Safety, and Health (ES&H), and Performance Assurance (Quality Assurance and Facility Evaluation Board) organizations, among others, report directly to the president and chief operating officer of FDH and provide oversight to ensure institutional safety provisions are implemented.

17 3 2 Organizational Responsibilities

A summary of DOE roles and the responsibilities of DOE organizations is presented in Sections 17 3 2 1 and 17 3 2 2. The roles of FDH as the managing contractor, the major subcontractors as operations contractors, and the enterprise companies as support organizations to FDH and other onsite contractors are summarized in Sections 17 3 2 3 through 17 3 2 7. Performance assessments and review organizations are summarized in Section 17 3 2 8.

17 3 2 1 U S Department of Energy, Headquarters The DOE, Headquarters (DOE-HQ) general responsibility is to provide funding and direction as required for the SNF Project. Specific responsibilities are provided in the following subsections.

17 3 2 1 1 U S Department of Energy, Headquarters, Assistant Secretary for Environment, Safety, and Health Assistant Secretary for Environment, Safety and Health responsibilities include the following:

- Provide technical support and independent oversight
- Provide support for the National Environmental Policy Act of 1969 (NEPA) documentation
- Participate in the Regulatory Requirements Team (RRT) for developing the project regulatory strategy and safety analysis report (SAR)
- Provide administrative support to the Independent Review Panel (IRP)

Roles and responsibilities of the RRT and IRP are provided in Section 17 3 2 8 4 and 17 3 2 8 5.

17 3 2 1 2 U S Department of Energy, Headquarters, Assistant Secretary for Environmental Management Assistant Secretary for Environmental Management responsibilities include the following:

- Establish overall mission objectives and top-level functional requirements
- Provide program oversight guidance and coordination with other DOE-HQ organizations
- Provide liaison with the national SNF program
• Provide liaison with the Office of Civilian Radioactive Waste Management program

• Participate as a member of the RRT

173213 U.S. Department of Energy, Headquarters, Office of the Associate Deputy Secretary for Field Management Office of the Associate Deputy Secretary for Field Management responsibilities include the following:

• Conduct independent cost estimates for the SNF Project as appropriate

• Participate in the validation of the SNF Project as part of the funding process

17322 U.S. Department of Energy, Richland Operations Office DOE-RL responsibilities are provided below

173221 U.S. Department of Energy, Spent Nuclear Fuel Project Division
Responsibilities for the management of the SNF Project are assigned through the deputy manager for site transition to the DOE Office of Spent Nuclear Fuels (SFO). The director of the SFO is the project manager and will conduct the project in accordance with DOE Order 4700.1, Project Management System.

The SFO organization is consistent with a projectized organization having approximately two-thirds of the staff matrixed from line organizations. The DOE SFO is responsible for the overall management, administration, performance, and operations and maintenance activities of the SNF Project. The DOE SFO ensures that the required levels of quality, safety, and environmental compliance are achieved within established technical, schedule, and cost baselines. Specific responsibilities and authorities include, but are not limited to, the following activities:

• Provide review and approval of SNF Project baseline documents

• Assume overall responsibility for design, construction, startup, and operation of systems and facilities within the scope of the SNF Project through direction to the contractor participants

• Monitor and maintain overview of project activities, cost and schedule status and technical baseline compliance to ensure project performance expectations, quality, cost, and schedule objectives are met

• Provide DOE interface with the DOE-ID/EM-67 National Spent Nuclear Fuel Program

• Provide and coordinate review and approval of required environmental and safety documentation

• Ensure quality, safety, and environmental requirements are applied
- Review and submit budget documents in accordance with the DOE budget cycle
- Interface with stakeholders
- Interface with the IRP

**17.3.2.2 U.S. Department of Energy, Office of Spent Nuclear Fuels Subproject Manager** The DOE SFO subproject manager is responsible for the management and administration of the SNF subproject activities, as well as for assessment of subproject performance and maintenance of subproject baselines. Specific responsibilities and authorities of the DOE SFO subproject manager include the following:

- Provide review and approval of SNF subproject documents including required environmental and safety documentation
- Approve overall scope, cost, and schedule baselines for the subprojects
- Monitor and maintain overview of subproject activities, cost, schedules, and status to ensure subproject performance, quality, cost, and schedule objectives are met
- Issue project authorizations for expense and capital funding as required by DOE-RL
- Release definitive design for construction and performance specification for design-and-build contracts

**17.3.2.3 Fluor Daniel Hanford, Incorporated** As management contractor, FDH is ultimately responsible for (1) contract performance, which includes protecting the public, workers, and environment from hazards associated with Hanford Site operations; (2) providing a focal point for interaction with DOE and the stakeholders; and (3) ES&H and Performance Assurance oversight. Figures 17-1 and 17-2 identify the organizations and organizational responsibilities within FDH that interface with SNF Project operations.

The FDH organization includes a Vice President and Project Director responsible for integrating the SNF Project into the overall Hanford Site organization. The SNF Project Vice President and Project Director provides leadership, oversight, direction, and control and employs the best commercial practices to the subcontractors in the execution of the SNF Project. The Vice President and Project Director monitors and is accountable for FDH subcontractor performance to ensure technical, cost, and schedule baseline conformance, integrating interfaces, evaluating and forecasting, and reporting performance. The reporting relationship of the SNF Project Vice President to the FDH President and Chief Executive Officer is shown in Figures 17-1, 17-2, and 17-3.

**17.3.2.3.1 Environment, Safety, and Health Organization** The ES&H organization reports to the highest operating levels in FDH to ensure coordinated development, incorporation, and implementation of environmental, safety, and health requirements into sitewide programs.
The environmental, safety, and health requirements are incorporated, as appropriate, into specific environmental, safety, and health programs associated with the SNF Project. The ES&H organization provides support to the SNF Project in the following areas:

- Occupational safety and health
- Emergency preparedness
- Nuclear safety and work controls
- Criticality safety
- Environment protection and regulation
- Radiation protection
- Planning and evaluation

Details of roles and responsibilities are provided in HNF-MP-003, *Integrated Environment Safety and Health Management System Plan*.

**17.3.2.3.2 Performance Assurance Organization** The FDH Performance Assurance organization is in place to develop sitewide quality assurance programs and policies and to communicate those programs and policies to the SNF Project. The Performance Assurance organization also reports to the highest operating levels in FDH. The purpose of FDH quality assurance programs is to establish an integrated approach to quality that ensures performance of all Hanford Site activities meets contract expectations. The quality assurance support function includes the following areas:

- Quality assurance programs
- Quality assurance surveillance
- Acceptance inspection
- Quality assurance technical support
- Performance assurance
- Codes, standards, and corrective action management

Details of roles and responsibilities are provided in HNF-MP-001 *Management and Integration Plan*.

The FDH Facility Evaluation Board and Independent Program Assessment organizations reside in the Performance Assurance organization to provide oversight activities including audits, surveillances, assessments, and appraisals of organizations performing environmental or quality-affecting activities.

**17.3.2.3.3 Training Organization** The training organization establishes the standards for all PHMC Team training to ensure that all training programs meet DOE-RL training requirements and to prepare the workforce to perform work safely and effectively. These standards, in conjunction with implementing procedures and supporting documents that govern training under the PHMC, establish a graded, systematic approach to training that is designed to ensure that the
qualifcation and training program meets DOE and federal requirements. The training organization provides support to the SNF Project in the following areas:

- Training programs/standards
- Training business operations
- Management information systems
- Training integration

Details of roles and responsibilities are provided in HNF-MP-001.

17.3.2.4 Spent Nuclear Fuel Project The SNF Project has responsibility for the design, construction, and operation of the SNF Project facilities. The SNF Project is the design authority and maintenance and operations contractor and has primary responsibility for executing the project mission. This includes defining systems through systems engineering, managing the subprojects and programs, providing technical direction to the design agents (architect-engineers) and the construction managers, reviewing and approving products and activities, and ensuring that requirements are met.

The SNF Project Vice President and Project Director has designed an organization and structure (Figure 17-3) combined of special projects and traditional work functions that provides management and technical support to the tasks of the SNF Project. The combination of FDH, SNF Project, SNF Project facilities, enterprise companies, and other onsite organizations ensures that responsibilities, resources, technical expertise, and management involvement are matched for safe, effective, and efficient accomplishment of the SNF Project mission.

The SNF Project organizations will transition from a design and construction, surveillance and maintenance (safe and stable storage of SNF in the K Basins) organization to a processing organization that will be involved with removing SNF from the K Basins, transporting the SNF for cold vacuum drying in the CVDF, and storing the SNF in the CSB.

The mission of the SNF Project is provided in HNF-3552, *Spent Nuclear Fuel Project Execution Plan*, and summarized below:

- Stewardship of applicable 100 K Area facilities until all of the SNF, debris, and sludge has been removed and turned over to the facility decontamination and decommissioning organization (Bechtel Hanford, Incorporated, currently has responsibility for portions of the K Area [BHI-00888]).
- Characterization of the K Basins SNF and sludge.
- Retrieval of all K Basins SNF for packaging and transportation to interim storage (managed as a subproject of the SNF Project).
- Removal and disposition of K Basins sludge and debris (managed as a subproject of the SNF Project).
- Treatment of the K Basins water until the SNF, debris, and sludge is removed
- Design and construction of the CVDF, including the fuel retrieval system, multi-canister overpack, and K Basin modifications, and of the CSB, including receiving and storage facilities perimeter fence, associated roadways and extension of Site utilities
- Operation of the CSB until all SNF from the K Basins is stored in a dry and stable configuration awaiting final disposition
- Stewardship of SNF from locations on the Hanford Site other than the K Basins that will be stored at the ISA adjacent to the CSB
- Other uses of the CVDF, which could include temporary storage of equipment such as the sludge removal system transport package
- Overall compliance with SNF Project quality assurance requirements and the proper delegation of that responsibility
- Development and maintenance of SNF Project-specific implementing documents that translate requirements into work processes
- Oversight of the implementation of specific safety program responsibilities to clearly demonstrate that safety is the first consideration for every activity, this includes responsibilities for accident investigation and corrective actions, worksite inspections and hazards analysis, conduct of safety meetings and prejob briefings, and employee safety training
- Design, construction, and operation of interim storage and stabilization facilities (near-term, interim, and stabilization) until turnover to an operations organization as described in this section

The SNF Project organizational structure is provided in Figure 17-3. This organization provides support to each of the SNF Project facilities. Responsibilities of the organizations reporting to the SNF Project Vice President and Project Director include, but are not limited to, the following

- Fulfilling the SNF Project mission
- Ensuring that design, construction, and operation meet all applicable Federal, state, and local laws and regulations and company standards and procedures
- Maintaining close liaison with DOE, the Defense Nuclear Facility Safety Board, and other external organizations
Establishing a management assessment process and ensuring management assessments involve senior management in determining the effectiveness of each organization and supporting organization.

- Ensuring the achievement of quality
- Implementing full compliance with the quality assurance program plan (QAPP)
- Ensuring QAPP compliance by outside SNF Project participants

17 3 2 4 1 Spent Nuclear Fuel Project Management SNF Project management is committed to providing for the safety and health of the workers and the public and protecting the environment during design, construction, and operation of each SNF Project facility. The responsibility for safety and health practices rests with the line organization. Responsibilities of the SNF Project primary organizations are presented below, the organizational structure is presented in Figure 17-3. Specifics on the organizational structure for K Basins are presented in Chapter 17 of WHC-SD-WM-SAR-062, K Basins Final Safety Analysis Report.

The SNF Project Vice President and Project Director is responsible for all aspects of project execution including safety, health, and environmental issues. Example responsibilities include the following:

- Operate the SNF Project facilities within the safety authorization basis and ensure safe operation of existing wet storage facilities and cost-effective transfer of SNF to a new dry storage facility by providing overall management direction to personnel performing the following functions: operations, maintenance, engineering, construction, safety and health, environmental protection, quality assurance, radiological controls, planning, contracts, procurement, training, scheduling, occupational safety and industrial hygiene, and others
- Direct the integration of multiple subprojects to successfully achieve project goals within schedule and budget constraints
- Develop and oversee the implementation of corrective measures to resolve complex technical and management issues threatening timely completion of the project
- Provide leadership and guidance for occupational safety and industrial hygiene issues. This includes providing leadership for implementation of the Voluntary Protection Program (VPP) and oversight of project construction activities
- Integrate project activities in the PHMC by facilitating effective communications and resolving issues among projects, ES&H, Performance Assurance, FDH subcontractors, PHMC support organizations, management, and DOE-RL
• Lead the SNF Project-specific execution of life-cycle baseline, multiyear, and current fiscal-year plans, integrate subcontractor activities into the planning process, and ensure project integration of these activities from a scope, schedule, interface, and resource perspective

• Interface with DOE-RL regarding incorporation of regulatory commitments and other requirements

• Use dedicated staff support, matrixed from FDH central support (e.g., ES&H and Performance Assurance) to perform verification activities and to formally assess performance

• Manage the overall execution of the SNF Project

17 3 2 4 2 Operations The SNF Project operations organizations (K East–K West Fuel Removal, SNF Transfer–CVDF, CSB SNF Waste Storage, Sludge Debris, Water Cleanup, and Other Site SNF Disposition shown in Figures 17-3, 17-4 and 17-5) manage and direct SNF Project operational activities in a safe, economic, and environmentally sound manner. All activities, including facility operations and maintenance, new facility startup and testing, handling and storage of SNF, and operational support functions (e.g., training, radiological control), are conducted in compliance with DOE contractual orders and applicable Federal, state, and local laws. Functional responsibilities are provided below:

Operations Operations successfully and cost-effectively executes the review process, start-up and ramp-up activities. Operations ensures that workers' safety and health are maintained at a high level of awareness and performance. Other specific responsibilities include the following:

• Operate the SNF Project facilities within the safety authorization basis

• Provide safe and cost-effective interim storage, conditioning, and relocation of SNF from 105K East and 105K West basins to dry, passive, interim storage within the CSB. Ensure activities are managed in accordance with applicable laws, regulations, and contractual obligations

• Develop and monitor performance to SNF Project goals, objectives, milestones, and assigned company performance agreements. Perform required self-assessment, and complete corrective actions with the goal of continuous improvement

• Ensure the development and preparation of schedules and financial documents containing program plans, budgets, and other accountability factors for the successful conduct of the SNF Project mission

• Provide operational input and review for all operational activities planned for the duration of the SNF Project
Execute effective integration and planning of all work activities of each SNF Project facility

Review facility design for operability, maintainability, and a defined safety basis that can be implemented in the field

Establish and maintain effective working interfaces with supporting organizations and act as liaison for Operations

Develop and coordinate initial staffing for operation of the facility and systems

Review preoperational test procedures

Designate representatives to support the Test Review Board, as required

Participate in the operational readiness review (ORR) process

Manage all basin operational activities to meet requirements on schedule and within budget

Maintain standard of performance for formal conduct of operations in accordance with DOE Order 5480 19 as defined in the conduct of operations applicability matrix

Provide surveillance data for all systems, conduct analysis and limited retention of data, and maintain up-to-date status of all facilities and equipment

Establish and coordinate the priorities for production and maintenance activities

Perform operational waste handling and associated analyses

Provide administrative management of the following programs (1) occurrence reporting, (2) key control, (3) lock and tag, (4) event investigation, (5) lessons learned, and (6) operator aids

Establish and maintain excellence in housekeeping

Startup Integration  Startup within Startup Integration (Figure 17-6) develops coordinates, and implements the SNF Project preoperational test program and supports the operational test program for the SNF Project facilities. This function will be absorbed into Operations when the SNF Project facilities are declared to be operational. Specific responsibilities include the following:

Manage test planning and scheduling, work force allocation, and budgeting activities
- Review and recommend revisions to design and procurement specifications, facility acceptance tests, construction acceptance tests, and contractor submittals as they relate to startup and testing
- Support preoperational test specification preparation
- Prepare preoperational acceptance test procedures in accordance with applicable test specifications
- Establish requirements and implement a system for access and work control during startup testing
- Provide maintenance support for equipment and systems that are turned over to Startup
- Manage organizational interface issues between Startup and other groups
- Review and accept project deliverables, which include hardware, software, certified vendor information, drawings, specifications, as-built information, test and inspection data, operation and maintenance procedures and manuals, and any other documentation related to safe and reliable operation of the systems, structures, and components (SSCs)
- Direct performance of preoperational acceptance testing and facilitate test deficiency resolution
- Assist in field training and operational "dry runs" to ensure that operators, procedures, and equipment are in a final state of readiness
- Coordinate control of test equipment, instrumentation, and other materials required to support testing activities
- Ensure special nuclear material accountability is maintained by establishing and maintaining custodial ownership of the special nuclear material
- Ensure all waste control activities for Operations meets the requirements of DOE orders along with applicable Federal, state, and local laws

**Operational Readiness Review**  The SNF ORR function (Figure 17-6) within Startup Integration directs and coordinates all key activities to successfully complete a management self-assessment for operational readiness, independent ORR, and DOE ORR in accordance with Section 1721 of the S/RID (HNF-SD-SNF-RD-001) and DOE O 425 1 for the SNF Project facilities. This includes overseeing the financial and schedule activities, updating the Startup Notification Report to FDH, and ORR records and consultational reviews related to ORRs. The
ORR function also provides oversight and assistance for SNF Project facility readiness assessments. Specific responsibilities include the following:

- Develop, maintain, and update the management self-assessment plan for operational readiness that will meet the requirements of DOE O 425 1, DOE C 425 1, CRD Startup and Restart of Nuclear Facilities, RLID 425 1, Startup and Restart of Nuclear Facilities, and DOE-STD-3006-95, Planning and Conduct of Operational Readiness Reviews (ORR).

- Develop and issue necessary plans of action to FDH and DOE-RL for all CSB activities as required by DOE O 425 1.

- Develop necessary plans, desk instructions and procedures to define and control project readiness determinations, management self-assessment, and ORR process.

- Provide interface with the senior ORR advisor from the DOE-RL staff (if assigned) the independent ORR chair, and the FDH organization responsible for Site ORR coordination, as well as those responsible for project-specific ORR activities.

- Conduct orientation briefings, with assistance from the SNF Project Training and Support Services personnel for all SNF Project staff associated with the management self-assessment or ORR activities.

- Provide guidance and on-the-job training for line managers performing management self-assessment activities for operational readiness and any management self-assessment validation activities to ensure that consistency and quality management self-assessment reviews for operational readiness take place.

- Provide 100% review of all management self-assessment appraisal forms for operational readiness for accuracy and completeness. This includes field assessment checks of select appraisals.

- Establish metrics from management self-assessment and ORR progress, track on a routine basis for trends that may need corrections, recommend changes based on results.

- Routinely monitor activities necessary to ensure appropriate implementation of the management self-assessment plan for operational readiness.

- Ensure implementation (administratively and through field verifications) of requirements from S/RLD assigned to the ORR group.
Procedures  Procedures development (Figure 17-4) includes startup project procedures, technical procedures, administrative support and S/RID coordination. Specific responsibilities include the following:

- Provide direction and maintain responsibility and control for SNF Project technical and administrative procedures in accordance with Section 10 of the S/RID (HNF-SD-SNF-RD-001), including development, control, and issuance, interact with other SNF Project organizations to ensure procedures comply with applicable standards and requirements.

- Provide procedure writers for the development, implementation and maintenance of the SNF Project technical procedures system (e.g., normal operating procedures, fuel handling procedures).

- Implement conduct of operations guidelines and ensure that activities for the training and procedures group are accomplished in a manner that supports procedure compliance, best management practices, and enhances employees' work environment.

- Ensure training and procedures are in compliance with Project Hanford Management System requirements (i.e., providing identified SNF Project functional area owner and lead reviews correspondent to the functional areas of the Project Hanford Management System), review and update SNF Project administrative and technical procedures to ensure compliance and clear guidance is maintained.

- Maintain SNF Project shared-area file server and electronic notification of changes to server for various SNF Project documentation (i.e., policies, charters, and administrative and technical procedures and documents).

- Maintain S/RIDs self-assessment database, provide data sorts for performance indicator charts and various other tasks as requested.

Training  Training includes operations training, maintenance training, and administrative support. Specific responsibilities include the following:

- Manage and direct activities related to SNF Project administrative and/or technical procedure development and the development of a comprehensive technical training program (see Chapter 12 for additional details on training and procedure development).

- Provide management of a comprehensive technical training program of appropriate formality and rigor to ensure that personnel are properly trained for the safe and efficient operation of all SNF Project activities.

- Provide development and instruction of technical training packages and provide scheduling services for required training for SNF Project personnel.
• Administer a comprehensive training plan for guiding technical procedure
development and field implementation for facility systems, and operating practices
and routines, including a qualification/certification program as described in
Section 4.0 of the S/RID (HNF-SD-SNF-RD-001)

• Provide point-of-contact for SNF Project management for all training needs

17.3.2.4.3 Maintenance  The SNF Project Operations Maintenance organization
(Figure 17-7) manages and directs activities for maintenance of the SNF Project facilities
Specific responsibilities include the following

• Direct all maintenance and work control activities in strict compliance with
DOE Order 4330.4B, Maintenance Management Program, Topical Area,
"Maintenance " Ensure that all corrective and preventative maintenance is performed
supporting operations in a manner that reflects the highest commitment to safety,
total quality, and excellence in a timely and cost-effective manner, and in accordance
with established procedures

• Participate in the overview of labor activities, ensure that assignments are in
accordance with the Hanford Atomic Metal Trades Council agreement and that all
grievances and employee concerns are addressed promptly and adequately

• Provide maintenance resources (allocation and coordination) for general facilities
support functions in the assigned areas  This includes transportation services,
management of contract support efforts, and procurement of maintenance materials

• Ensure maintenance personnel are sufficiently trained to meet the requirements of the
work assigned

• Provide daily direction of maintenance work tasks where needed

• Provide work package planning and final package assembly for all job control system
work packages

• Assemble track status, and close out all job control system work packages ensuring
document retention requirements are followed

• Provide material control of tools, shop stock, staged material, and warehoused
equipment

• Manage and control the use of measuring and test equipment and provide reverse
traceability capabilities
Ensure that prejob briefings are held addressing all safety and radiological issues, setting the high standards and expectations for each individual taking part in the work.

Ensure timely investigations of reported off-normal plant events (e.g., events, unusual occurrences and Occupational Safety and Health Administration recordables) and lost work day injuries.

Establish an integrated approach to manage work activities.

Monitor performance versus the established schedule and budget estimates and take action to maintain schedule and budget commitments.

Identify and resolve barriers to complete work activities.

Identify and implement work planning, scheduling, and implementation process improvements.

17 3 2 4 4 Spent Nuclear Fuel Project Engineering  The SNF Project Engineering Manager and Chief Engineer has overall responsibility for establishing, implementing, and maintaining the engineering technical baseline, authorization basis, and nuclear safety regulatory compliance. As design authority for the SNF Project, the Chief Engineer defines requirements for and ensures the technical adequacy of all SNF Project facilities SSCs. This authority ensures that the initial release and changes to all documentation that affect functions, requirements, architecture, interfaces, operability, maintainability, and safety basis are technically sound and consistent with the approved authorization basis.

Responsibilities of the Manager and Chief Engineer position include the following:

- Perform the Chief Engineer function and provide interpretive authority for the resolution of technical issues.
- Establish and maintain an engineering program infrastructure consistent with the SNF Project mission scope and program logic.
- Direct technical integration and coordination to ensure that key SNF Project engineering goals are achieved.
- Lead and manage interfaces with customers, the public, and stakeholders in technical matters associated with the SNF Project mission.
Functional responsibilities of the Engineering organization are as follows

- Facility engineering
  - Provide day-to-day engineering support for the operations and maintenance of the K Basins, CVDF, CSB, and ISA facilities and systems
  - Configuration control of plant design documentation and engineering infrastructure
  - Provide cognizant facility systems engineers and supporting staff to perform responsibilities
  - Provide engineering expertise for plant walkdowns to ensure safe and effective implementation of maintenance upgrades and plant modifications
  - Develop and maintain in-depth knowledge of system design, performance, and current condition through regular system walkdowns, system performance reviews, and communications with SNF Project facility operations, maintenance, and engineering personnel
  - Provide engineering support for work package preparation, execution, and closeout
  - Design, recommend, select, and specify new plant equipment and modifications to plant equipment and systems to support the SNF Project facilities using engineering change notices
  - Provide the design of systems, components, and equipment required to support fuel removal activities, remove waste streams that could result in a potential insult to the environment, and provide and implement dose reduction for decontamination and decommissioning activities
  - Ensure all provided systems are operable within the existing safety authorization basis and technical safety requirements (TSRs), or support necessary updates to the SNF Project Final Safety Analysis Report (FSAR)
  - Support/maintain corrective and preventive maintenance of SSCs
  - Conduct the fire protection program
  - Resolve operations issues
  - Monitor instrumentation and alarms
• Systems engineering
  – Establish the engineering processes and monitor performance of engineering activities
  – Develop and maintain engineering procedures and maintain the configuration management plan
  – Assess performance of engineering activities relative to the approved procedures, the configuration management plan, and other applicable PHMC requirements

• Nuclear safety
  – Provide overall coordination and integration of the SNF Project regulatory matters, including nuclear safety regulations by interfacing with FDH, DOE, and other stakeholders
  – Prepare all safety analysis documents for the SNF Project facilities and perform independent reviews of activities documented with safety implications
  – Manage USQ process for SNF Project facilities

• Process engineering
  – Define process technical basis
  – Develop technical basis documents
  – Provide process validation and monitoring requirements for fuel storage processes
  – Develop thermal-hydraulic models of the multi-canister overpacks in process and storage that will support completion of the SNF Project FSAR and system designs
  – Produce and maintain process flow diagrams and process technical manuals and integrate cross-cutting calculations for subprojects
  – Manage characterization data

• Process control
  – Monitor the overall engineering process relative to technical basis, safety, and regulatory set points
- Provide answers to operational, safety, and quality assurance inquiries
- Maintain chemical and technology information to address questions during operational processing

• Technical integration
  - Provide design authorities for the SNF Project
  - Design authorities interface with cognizant engineers at each SNF Project facility to ensure a strong technical basis for operation, to maintain the SNF Project facility in conformance with the authorization basis, and to build a culture of excellence in engineering
  - Assess performance of engineering activities relative to the approved procedures, the configuration management plan, and other applicable PHMC requirements
  - Provide other design authority responsibilities

• Technical operations
  - Plan, direct, and complete all work necessary to ensure that SNF Project technical issues are identified and brought to closure such that project objectives are achieved
  - Work with managers throughout the SNF Project to facilitate identification of technical issues
  - Ensure that closure plans appropriate to specific issues are prepared
  - Ensure that closure criteria are developed and consensus on the criteria are achieved
  - Bring together and direct all engineering, subproject, or corporate resources necessary to achieve closure
  - Provide oversight of lower tier issues management
  - Integrate issue resolution into project schedules
  - Track and status progress for issue closure
17.3.2.4.5 Spent Nuclear Fuel Project Quality Assurance  The SNF Project Quality Assurance organization (Figure 17-3) develops and maintains the SNF Project QAPP, and interprets and sets quality assurance criteria for SNF Project organizations. The SNF Project line organizations have the overall responsibility for performing work activities within the established criteria. Specific responsibilities of the SNF Project Quality Assurance organization include the following:

- Interpret and approve quality assurance program requirements
- Identify quality problems, initiate, recommend, or provide solutions to quality problems, and verify those solutions are implemented and effective
- Perform oversight activities on workscope to verify implementation of quality assurance requirements
- Provide overview of project management's implementation of the QAPP and implementing procedures
- Review project documentation such as SARs, S/RIDs, and project management plans
- Prepare and support ORR activities to incorporate quality assurance requirements

17.3.2.4.6 Radiological Controls  The SNF Project Operations Radiological Control group (Figure 17-8) administers an occupational radiation safety program that is in full compliance with Section 110 of the S/RID (HNF-SD-SNF-RD-001) and Title 10, Code of Federal Regulations, Part 835 “Occupational Radiation Protection” (10 CFR 835), and in a manner commensurate with HSRCM-1 Hanford Site Radiological Control Manual, for the SNF Project. Specific responsibilities include the following:

- Manage an occupational radiation safety program that is in full compliance with 10 CFR 835 and in a manner commensurate with HSRCM-1
- Perform radiological surveillance and monitoring as specified by 10 CFR 835, Subpart E, Article 835 401
- Administer ALARA (as low as reasonably achievable) Program as specified by 10 CFR 835, Subpart B, Section 835 101 C and 10 CFR 835 Subpart K, including program implementation, design reviews and work control processes pertaining to radiological work and facilities
- Issue dosimetry in support of 10 CFR Subpart E, Article 835 402. In this context, "field dosimetry" applies to job-specific dosimetry only; it does not apply to dosimeters that are issued by SNF Project dosimetry

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- Identify the need for, and location of, area monitors and air sampling equipment as specified by 10 CFR 835, Subpart E, Article 835 403. Radiological Control personnel obtain readings from these instruments for the purpose of performing occupational monitoring.

- Perform radioactive contamination monitoring as specified by 10 CFR 835, Subpart E, Article 835 404.

- Provide radiological entry control and enforce SNF Project-specific entry control criteria that are consistent with 10 CFR 835, Subpart F, Articles 835 501 and 835 502.

- Establish SNF Project-specific radiological posting criteria that are consistent with 10 CFR 835, Subpart G.

- Administer processing of legal record documents (generated by Radiological Control personnel) that document radiological monitoring, surveillance, dosimetry issue, and radiological access control activities.

- Ensure that health physics technicians receive sufficient training as defined by 10 CFR 835, Subpart J, Article 835 903.

- Prepare radiation work permits that specify adequate protective measures for radiation workers as defined by DOE/EH-0256T, *U.S. Department of Energy Radiological Control Manual*.

- Perform as the SNF Project interpretive authority for 10 CFR 835 and DOE/EH-0256T.

- Maintain a reporting program that identifies radiological issues and alerts SNF Project line management to compliance issues associated with 10 CFR 835 and/or applicable DOE orders governing radiological safety.

- Provide quick turnaround radiological-count-room services, according to 10 CFR 835, HSRCM-1, DOE Order 5400 5, *Radiation Protection of the Public and the Environment*, and SNF Project quality assurance protocols for (1) radiological control samples (air, smear soil, and miscellaneous items) and (2) radiological release of material and equipment.

**17 3 2 4 7 Spent Nuclear Fuel Project Safety, Health and Emergency Planning**  The SNF Project Safety, Health and Emergency Planning organization (Figure 17-3) manages safety and health, chemical management, and emergency planning. Specific responsibilities include the following:

- Provide leadership and guidance for emergency planning issues.
- Maintain, review, revise, and implement the building emergency plan and applicable facility-specific emergency response procedures

- Provide leadership and guidance for chemical management

- Develop and administer programs to control and minimize health and safety hazards and to promote continuous improvement in safety and health performance

- Provide and manage resources to design and implement occupational safety and health programs

- Ensure that applicable requirements are integrated into the occupational safety and health program and the procedures and standards of other applicable organizations

17.3.2.4.8 Performance Improvement and Regulatory Services  The SNF Project Performance Improvement and Regulatory Services organization (Figure 17-3) manages environmental protection, corrective action program, self-assessment management systems, performance indicators, requirements flowdown and S/RIDs, and regulatory interface and technical support. Specific responsibilities include the following:

- Prepare and ensure timely approval and implementation of environmental permits for the project. Ensure other project and site-side environmental initiatives are implemented and act as point-of-contact for environmental issues.

- Manage the SNF Project corrective action program and interface with the Site corrective action program.

- Ensure problems and issues are evaluated and reported in accordance with the Price Anderson Amendments Act of 1988.

- Provide a management assessment program that meets applicable criteria.

- Assist management with the selection, design, development, and implementation of the performance indicators.

- Monitor and assist management with maintaining an accurate set of requirements with the SNF Project S/RIDs.

17.3.2.5 Other Project Hanford Management Contract Team Companies  The following list includes the major subcontractors and a summary of their roles (HNF-MP-599):

- Numatec Hanford Corporation, which is responsible for engineering and technology support.
B&W Hanford Company, which is responsible for the Facility Stabilization Project and the Advanced Reactors Transition Project

Waste Management Federal Services of Hanford, Incorporated, which is responsible for the Waste Management Project and for environmental management and permitting

Lockheed Martin Hanford Corporation, which is responsible for the Tank Waste Remediation System Project and for providing support to FDH site systems engineering

DynCorp Tri-Cities Services, Incorporated, which is responsible for infrastructure services, including facility maintenance, site services, real estate, and property management and emergency preparedness

17 3 2 6 Enterprise Companies that Interface with the Spent Nuclear Fuel Project - Several enterprise companies provide support to SNF Project organizations as matrixed functions. The interfaces between these support companies and the applicable SNF Project organization are controlled by the SNF Project for matters related to cost, schedule, priorities, and safety management. These enterprise companies include:

- Fluor Daniel Northwest, Incorporated, which provides architect/engineering and construction management to the major subcontractors for project and/or task activities

- Lockheed Martin Services, Incorporated, which provides information resource management services and support, including telecommunications and network engineering, information systems, production computing media services, document control, and records management

- Waste Management Federal Services, Incorporated, Northwest Operations, which provides environmental management technical support to the major subcontractors and other services such as integrated pest management, groundwater well services and sampling, and transportation, packaging, and hazardous shipping certification and training

- COGEMA Engineering Corporation, which provides equipment engineering, process engineering, and professional services

- Protection Technology Hanford, which provides support services for the safeguards and security workscope, including special nuclear material control, accountability, physical security, and information security
17 3 2 6 1 Hanford Fire Department  The Hanford Fire Department's jurisdiction encompasses all of the Hanford Site. The Department is managed by DynCorp Tri-Cities Services. The Hanford Fire Department is responsible for providing the following fire protection response functions:

- Maintain command of emergency response forces to control and terminate fire-related incidents, provide emergency medical patient care, and act as the incident command agency for emergency incidents involving hazardous materials.
- Conduct functional testing and maintenance of fire protection self-contained breathing apparatus at all Hanford Site facilities.
- Maintain an active fire prevention program through facility tours and inspections of all flammable and reactive waste sites.
- Maintain a highly trained emergency response team certified under a three-year, Washington State-approved, fire fighter apprenticeship program.
- Maintain a modern fleet of emergency response vehicles consisting of pumper trucks, aerial ladder units, modular ambulances, command vehicles, wild land brush tankers, hazardous materials response vehicles, and specialized equipment designed to deal with the unique requirements of the Hanford Site.

The Hanford Fire Department works closely with the SNF Project Safety and Health and Quality Assurance organizations to support related inspection compliance, and design-review activities. Chapters 11 0 and 15 0 address the fire protection issues that apply to the safety of SNF Project facility operations.

17 3 2 6 2 Procurement and Materials Management  Procurement and Materials Management within the Business Systems group is managed by FDH with support from Fluor Daniel Northwest, Incorporated. The responsibilities of the Procurement and Materials Management organizations include the following:

- Procuring goods and services that meet established standards and requirements in accordance with approved procedures and processes with graded levels of management controls.
- Evaluating prospective suppliers against specified criteria.
- Ensuring that approved suppliers continue to provide acceptable items and services.
- Coordinating inspection and testing of specified items, services, and processes to ensure that established acceptance and performance criteria are used.
17 3 2 6 3  Transportation and Packaging Services  The transportation and packaging services organization, managed by Waste Management Federal Services of Hanford, provides sitewide policies and procedures governing hazardous materials transportation and packaging, qualification requirements and shipper training, preparation and maintenance of required shipping container safety documentation, hazardous materials packaging, design procurement support, analysis and testing, field support to the DOE-HQ transportation management program, and all other transportation management activities.

17 3 2 7  Other Onsite Contractors  Several other onsite contractors provide support to the SNF Project. The responsibilities of these contractors are discussed in the following subsections.

17 3 2 7 1  Pacific Northwest National Laboratory  The Pacific Northwest National Laboratory has responsibility for providing personnel monitoring which includes personnel dosimetry monitoring, internal dosimetry monitoring, dosimetry record management, and dose evaluation in emergency cases. The laboratory provides support to the DOE-RL and its contractors for the Radiation Instrumentation Monitoring Program. This service includes the evaluation of new instruments, maintenance of emergency response units and instruments, and the routine calibration of radiation survey instruments. The organization also performs extensive Hanford Site characterization work. Pacific Northwest National Laboratory is available to provide additional research and development support as well as review, consultation and report preparation support.

17 3 2 7 2  Site Medical Contractor  In accordance with DOE Order 5480.8A, Contractor Occupational Medical Program, the Hanford Site medical contractor serves two functions for the subprojects: (1) maintaining the medical monitoring programs for employees in compliance with Occupational Safety and Health Administration standards and DOE orders and (2) monitoring staff safety performance through reports of on-the-job injuries. The medical monitoring responsibilities include performing preplacement, periodic, return-to-work, and fitness-for-duty examinations for all workers, performing physical examinations for workers who handle hazardous waste, maintaining medical records, providing management consulting, managing the employee occupational exposure database, and managing the material safety data sheet database.

17 3 2 8  Performance Assessment and Review Organizations  As required by 10 CFR 830.120, the performance assessment and review program provides for the measurement of item and service quality, requirements compliance and work performance and promotes improvement. The assessment process incorporates a performance-based approach with emphasis on the results of work processes and compliance with requirements. (See Chapter 14.0 for additional details on the management and independent assessment program including safety issue discovery, communication management, and resolution of issues.)

Several internal and external organizations provide independent performance-based and compliance-based reviews of the SNF Project to applicable requirements. These organizations are described in the following subsections. In addition, management self-assessments are required by 10 CFR 830.120 and are performed as described in Section 17 4.1. An administrative system for
tracking action items identified as a result of internal and external audits, appraisals, evaluations, self-assessments and project schedules is described in SNF Project implementing procedures.

17.3.2.8.1 Quality Assurance Public safety, onsite worker safety, and protection of the environment are overriding considerations for the SNF Project. The commitment to safety by the SNF Project management dictates management and verification activities to ensure safety and environmental considerations are reflected in the design, procurement, construction, and operation of the project facilities in accordance with 10 CFR 830. For purposes of this document, quality assurance is all of those planned and systematic actions necessary to provide adequate confidence that an SSC will perform satisfactorily in service. Periodic inspection of SSCs in accordance with DOE Order 4330.4B will be performed to determine whether deterioration is taking place and to address technical obsolescence that threatens performance, safety, or facility preservation.

The SNF Project QAPP will meet the applicable requirements identified in Section 14.2. Assessments will be conducted to verify that these quality requirements are being met. The organizations that are responsible for providing quality assurance support are identified in Section 17.3.1. The methodology used to determine the quality assurance requirements and procedural controls selected for specific activities includes the following:

- Identifying items and activities that are candidates for procedural control by the quality assurance program

- Selecting the quality assurance requirements (and their degree of application) that are appropriate for each candidate item, component, system, structure, and activity based on the following considerations:
  - Impact of malfunction or failure on safety, health, and environmental protection
  - Design and fabrication complexity
  - Reliability of process
  - Reproducibility of results
  - Uniqueness of product
  - Degree of functional product demonstration
  - Quality history and degree of standardization
  - Impact on cost and schedule to replace in the event of failure
  - Environmental impacts due to failure
- Degree to which functional compliance can be demonstrated by inspection or test
- Necessity for special controls and surveillance over processes
- Significance to any applicable licensing process

The type of independent assessment performed and the frequency with which an assessment is performed will be based on the status, complexity, and importance of the activity or process being assessed, and the past performance of the activity or process. Independent assessments will be conducted by staff members who are technically qualified and knowledgeable in the activity or process being assessed and who have sufficient authority and freedom from the line organizations to carry out their responsibilities. An independent assessment will identify:

- Work performance and process effectiveness
- Abnormal performance and potential problems
- Improvement opportunities

SNF Project contractor participants, including subcontractors and suppliers, are required to implement quality assurance programs commensurate with the contract document requirements. Work is to be performed by contractors having DOE-approved quality assurance programs. Contractors or suppliers providing services for the SNF Project that do not have a DOE-approved QAPP will develop a QAPP meeting the requirements of the SNF Project specific to their area of responsibility. As an alternative, they may elect to use the SNF Project QAPP. QAPPs developed by contractors or suppliers are submitted to the SNF Project for review and approval.

The architect-engineer, construction management, and construction subcontractor organizations are each required by the QAPPs to identify the quality assurance position within their organizations that is responsible for the establishment and implementation of their QAPP. The following characteristics are required of this position:

- An organizational position at the same or higher organizational level as the highest line manager responsible for performing activities affecting quality
- Knowledge and experience in the areas of quality assurance and management
- Authority and responsibility to verify the adequacy and implementation effectiveness of the QAPP and subtier QAPPs
- No other duties or responsibilities unrelated to quality assurance that could prevent full attention to quality assurance program matters
- Sufficient freedom from cost and schedule considerations when such considerations could impact quality or safety
- Access to senior management and management at the next higher organizational level to identify and obtain resolution of quality concerns

- Ability to review and approve the QAPP and subsequent revisions and interpret quality assurance requirements

17 3 2 8 2 Safety Review Board The primary functional responsibility of the Safety Review Board (SRB) is to provide a formal review and approval process for safety basis documents to ensure adherence to the requirements of DOE orders and standards.

Safety basis documents assure the DOE that a nuclear facility can operate safely and be environmentally compliant. They include, as a minimum, ORRs, Plan of Actions/Memorandums of Agreement, SARs, TSRs, fire hazard analyses, criticality safety evaluation reports, and SARs for packaging.

SRB members are appointed by the SNF Project Director and include individuals who serve a minimum one-year term.

An SRB quorum consists of the chairman or vice chairman and any two members of the SRB. No substitutes are allowed in calculating the quorum (unless formally assigned in writing by the SRB member indicating a permanent alternate). However, managers are encouraged to send a representative as a nonvoting member for discussion purposes if they cannot attend.

All SRB members must be familiar with key authorization basis documents and applicable DOE orders. The SNF Project SRB members shall have broad-based knowledge of SNF Project operations, facility design basis, project schedule, and other requirements to effectively review and approve SNF Project safety basis documents. Each member should be knowledgeable in a technical field and familiar with the current SNF Project baseline and path forward planning.

The cognizant manager of a safety basis document shall advise the committee chairman of the need for a committee review and shall prepare a presentation for the meeting. The SRB will perform a documented review of authorization basis documents and changes. SRB-approved authorization basis documents that require DOE-RL approval are submitted to FDH for submittal to DOE-RL.

17 3 2 8 3 Facility Evaluation Board Facility-independent assessments are performed by the Facility Evaluation Board, which is not associated with the activity or process being assessed. Responsibilities include the following:

- Establish, implement, and maintain an effective independent oversight program to assess operations and management effectiveness in meeting established environmental, safety, health, and quality assurance requirements, applicable regulations, and sound management practices.
- Provide facility managers and senior management of the SNF Project and subprojects with clear, objective, and well-documented assessments of strengths and weaknesses that affect performance to approved standards

- Ensure that independent assessments are conducted of programmatic or functional areas of environment, safety, health, and quality assurance when required by regulation or contract requirements

The scope of the independent assessments includes, but is not limited to, radiological controls, occupational safety and health, quality assurance, environmental programs, engineering, training, configuration management, maintenance, nuclear safety, fire protection, and operations.

17 3 2 8 4 Regulatory Requirements Team  An RRT has been established to review and approve selected regulatory requirements applicable to the new SNF Project facilities and fuel products removed from the K Basins that have an influence on the fuel acceptance criteria. The RRT also supports the safety review and authorization process for SNF Project requirements. The RRT as a body has membership thoroughly familiar with the DOE and U.S. Nuclear Regulatory Commission (NRC) regulatory requirements. The members of the team are selected for their knowledge base and are from the various organizations involved with the SNF Project DOE-RL, DOE-HQ (Office of Assistant Secretary for Environmental Management and Office of Assistant Secretary for Environmental Health) and the operating contractor (FDH). Membership also includes outside consultants selected for their knowledge. The RRT is impaneled to support the SNF Project in its mission and, as such, the RRT advises and reports to the SFO Project Director.

The organizational responsibilities are described in Regulatory Requirements Team (RRT) Charter (Sellers 1997).

17 3 2 8 5 Independent Review Panel  The regulatory policy for the SNF Project as stated in WHC-SD-SNF-SP-002, Spent Nuclear Fuel Project Regulatory Strategy, delineates DOE-RL's policy for achieving nuclear safety equivalence for the design, construction startup, and operation of new facilities, and the new processes for removal, handling, and conditioning of the SNF for transfer to the interim storage facility. An IRP has been established reporting to the Office of the Manager of DOE-RL, to provide advice and high-level oversight of the implementation of the regulatory policy. Detailed organizational responsibilities of the IRP are documented in RL-D96-007, Project Plan/Charter, Spent Fuel Project. To accomplish this function, the IRP will conduct interviews, review documents, and attend presentations conducted by the SNF Project to reach conclusions relative to the adequacy of implementation of the regulatory policy.

Specifically, the IRP will

- Review and comment on the SNF project regulatory strategy
- Evaluate and concur with the regulatory requirements document established by the project
- Evaluate and concur with the SAR and safety evaluation report
- Concur with the safety basis for final approval to operate the SNF Project facilities

The IRP consists of three members from outside the DOE complex, with broad technical background, selected for their stature, technical capability, and experience applicable to the SNF Project. The IRP conducts its oversight activity concurrently with the regulatory process. Periodic meetings of the IRP are conducted to enable the panel to keep abreast of SNF Project subproject activities. From time to time, DOE may identify further activities or areas that the IRP may be requested to review. IRP conclusions with respect to each of the chartered responsibilities are expected to be made within the time allotted by the subproject schedule. In interfacing with a subproject, the IRP's principal point of contact is DOE-RL's SFO director or the senior technical advisor to the assistant manager for waste management.


The SNF Project path forward requires approval of several key documents by DOE-RL before initiating operation of the new facilities referenced in the regulatory policy. Principal documents include the SARs prepared for the SNF Project facilities.

The DOE SFO and the Office of Assistant Manager for Engineering and Standards (AMS) collaborate to support the review and approval of key documents prepared by the contractor that are essential to the SNF Project. In addition to approving the SAR and issuing the safety evaluation reports, the AMS and SFO interact on a regular basis with the IRP to resolve technical issues in the SAR as they pertain to the implementation of the new regulatory policy. The DOE-RL SAR review team and the IRP jointly assess the effectiveness of the SFO in implementing the regulatory policy and the DOE orders that direct quality, safety, and health requirements that will be used for the project. The AMS also provides technical support to the SNF Project RRT. The Regulatory Strategy Document (WHC-SD-SNF-SP-002) contains the schedule for performing the SAR reviews.

1733 Staffing and Qualifications

Safe and effective design, construction, and operation of the SNF Project facilities depends upon a staff of qualified, competent personnel that conduct routine and long-range activities. Minimum education and experience requirements for SNF Project facility managers and technical personnel meet the requirements of DOE Order 5480 20A. Qualifications are met through education and related industry experience and through company-sponsored, job-specific training.

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Training and qualification requirements for SNF Project personnel are summarized in Chapter 12. Other subject-specific training requirements are
summarized in the applicable chapters of this SAR. In addition to meeting the SNF Project and subject-specific training requirements, each SNF Project employee must also successfully complete the following Hanford Site training programs:

- Hanford general employee training
- New employee safety orientation
- Appropriate facility orientation
- Appropriate facility building emergency plan

17.3.3.2 Minimum Staffing Requirements  DOE Order 5480 23 requires that the number of managers, engineers, operators, and support personnel assigned to a SNF Project facility is adequate to support safe design and construction activities and subsequent operation of the SNF Project facility. Abnormal plant conditions are considered when determining staffing assignments including on-shift or off-shift personnel. Establishment of minimum staffing levels for the various SNF Project facility positions is dependent on a job task analysis performed by management with considerations of normal and abnormal plant conditions, allocations for emergency response, monitoring functions, need for backups, availability of temporary staff from other organizations for peak workload periods, on-shift and off-shift personnel, and other factors. Minimum staffing requirements for safe operation are identified in the final TSRs.

17.3.3.3 Fitness for Duty  Fitness for duty involves an individual's ability to perform assigned tasks free from impairments caused by drug or alcohol abuse, emotional distress, or personal health problems. SNF Project management is committed to providing a work environment free of the negative influences of behaviors that jeopardize the work practices required for safe operations and quality services in support of DOE Order 5480 23. All employees will be fit for duty at their assigned tasks. Use, sale, or possession of illegal drugs or alcohol is prohibited on the Hanford Site. Employees who are found to be sellers, distributors, or repeated users of illegal drugs will be denied access to the facilities. To ensure this commitment is met, the SNF Project has an employee assistance program and has trained managers and supervisors to help them better recognize symptoms affecting fitness for duty.

17.4 SAFETY MANAGEMENT POLICIES AND PROGRAMS

The integrated environment, safety, and health management system (ISMS) (HNF-MP-003) establishes a single, defined safety and environmental management system that integrates ES&H requirements into the work planning and execution processes of the project facility effectively protecting the workers, public, and the environment. The ISMS is primarily based on the philosophies, principles, and requirements of DOE P 450 4, Safety Management System Policy, and the specification and guidance for environmental management systems (ANSI/ISO 14001).

The safety management system policy described in DOE P 450 4 is based on the following principles:

- Line management is responsible for safety.
clear roles and responsibilities are established
- Competence of staff is commensurate with responsibilities
- Priorities are balanced
- Applicable safety standards and requirements are identified and implemented
- Hazard controls are tailored to work being performed
- Operations authorizations

The safety management functions of DOE P 450 4 provide the necessary structure for any SNF Project facility work that could affect the workers, public, or the environment. These functions are:

- Scope of work is defined
- Hazards are analyzed
- Hazard controls are developed and implemented
- Work is performed within established controls
- Feedback and continuous improvement is part of the safety culture

The overall management system of each SNF Project facility includes organizational structure, planning activities, responsibilities, practices, procedures, processes, and resources for developing, implementing, achieving, reviewing, and maintaining the environmental management system consistent with the principles of ANSI/ISO 14001, Environmental Management Systems – Specification with Guidance for Use.

HNF-MP-003 contains a crosswalk so that a correlation can be identified for implementation of the elements of ANSI/ISO 14001 and the principles and functions of the ISMS and stipulates the ES&H requirements that will be met in the SNF Project facilities. The ES&H programs include environmental protection, occupational safety, fire protection, industrial hygiene, radiological control, nuclear safety, emergency preparedness and radioactive and hazardous waste management.

Communication and coordination of ES&H programs, including safety issue discovery, management, and resolution, are described in the following subsections and in various implementing procedures for the SNF Project facilities.

17.4.1 Safety Review and Performance Assessment

Safety performance is reviewed and assessed by independent organizations identified in Section 17.3.2.8. Management assessments are required by 10 CFR 830 120, “Quality Assurance Requirements” and are conducted by an organization manager to regularly assess the adequacy of those portions of the quality assurance program for which that manager is responsible to ensure its effective implementation (see Chapter 14.0). The management assessment process for the SNF Project is contained in SNF Project administrative procedures. A management oversight program has also been implemented to ensure high standards of performance in the areas of
In addition, documentation establishing the regulatory and safety basis for the major activities and facilities of the SNF Project will require review and approval by staff of the SNF Project and DOE. SNF Project administrative procedures define the process for preparation, revision, and distribution of the SAR and implementing TSRs and/or operating safety requirements. A discussion of these documents and the review process is provided in the following subsections.

17 4 1 1 Regulatory Policy DOE-RL has established a special regulatory approach to resolve safety concerns associated with the SNF presently stored in the K Basins and with its movement to the CSB for storage following conditioning in the CVDF. The DOE-RL regulatory policy, as contained in WHC-SD-SNF-SP-002, has been approved and will be applied to matters of nuclear safety (including radiological control issues) for new SNF Project facilities.

It is DOE-RL and DOE-HQ policy (Grumblly 1995) that the design and construction of new SNF Project facilities will achieve nuclear safety and radiation protection equivalence with comparable facilities licensed by the NRC. This will be accomplished by applying technical requirements based on those applied by the NRC to comparable licensed facilities or activities in addition to applicable DOE requirements. If cases arise where DOE requirements are more stringent than NRC requirements, then DOE requirements have been selected and applied. In cases where NRC requirements are more stringent than DOE requirements, NRC requirements have been selected and applied. Exceptions will be documented for approval by DOE and the RRT in accordance with DOE's policy.

The regulatory policy's primary focus is on design and construction issues and preparation for operation. The policy does not apply to environmental, Occupational Safety and Health Administration, chemical accident safety, and other nonnuclear or radiation safety issues, these will be covered elsewhere by DOE orders and statutory requirements. Similarly, life-of-facility oversight (e.g., operator training/performance assessment) applied by the NRC to their licensed facilities is not covered by this policy (WHC-SD-SNF-SP-002).

DOE has established this policy for the following reasons:

- To achieve in the design and construction of the SNF Project facilities a level of nuclear safety and radiation protection comparable to that of NRC-licensed nuclear facilities.
- To enhance public understanding and confidence in the safety of the new SNF Project facilities by following an enhanced regulatory strategy.

17 4 1 2 Industrial Safety and Hazardous Material Control Industrial safety and construction safety requirements are identified in DOE Order 5480 9A, Construction Project Safety and Health Management, DOE Order 5480 10, Contractor Industrial Hygiene Program,

These policies and procedures ensure that SNF Project employees are provided with adequate protection, training, and information about industrial safety, construction safety, hazardous waste, and emergency response actions to conduct SNF Project operations safely. These policies and procedures also apply to activities performed by organizations, contractors, and subcontractors supporting the SNF Project organization.

SNF Project implementing procedures reflect all the elements of the DOE VPP provided in DOE/EH-0433, U.S. Department of Energy Voluntary Protection Program and in HNF-MP-003:

- Management commitment
- Employee involvement
- Hazard prevention and control
- Worksite analysis
- Health and safety training

Safety training of SNF Project management and workers places emphasis on the elements of the VPP and workers are encouraged to actively participate in VPP initiatives.

SNF Project implementing procedures outline the rights and responsibilities of SNF Project management and employees (including managers and supervisors) and building managers and administrators. This procedure also contains the Master Safety Rules, with which all employees must comply, and a Workers Bill of Rights, which contains the rights of all employees pertaining to safety. Each SNF Project manager and supervisor is responsible for maintaining active involvement in safety issues. Responsibilities include the following:

- Incorporate specific safety and health responsibilities and requirements into formal job descriptions, clearly communicate them to the affected employees, and establish accountability through a program of performance measurements
- Hazard evaluation development
- Worksite inspections and assessments
- Safety meetings and committee functions
- Accident and near-miss investigation and corrective action planning
- Prejob briefings
Resolution of safety concerns

Worker representation in all facets of safety program development and review

Each SNF Project employee is responsible for recognizing and promptly notifying management of events and conditions that could have adverse safety or environmental implications. Employees are also responsible for obeying all oral and written instructions and procedures, conscientiously applying all health and safety training in daily activities, and properly using all prescribed personal protective equipment. In addition, SNF Project employees have the following specific responsibilities:

- Observe the Master Safety Rules
- Actively support programs designed to protect employees
- Stop work to prevent or control hazards considered to be an immediate threat
- Work defensively and be vigilant that coworkers do not put themselves at risk
- Submit ideas for safety improvements to management
- Report injuries and medically imposed work restrictions to their immediate managers

Performance requirements for reporting, investigating, and managing events that have safety or health significance and for complying with DOE Order 54841, Environmental Protection Safety and Health Protection Information Reporting Requirements, are presented in HNF-PRO-077, Reporting and Investigating Accidents. The requirements include reporting, investigating, and managing the following events:

- Occupational injuries and illnesses
- Government motor vehicle accidents
- Property damage incidents
- Near-miss incidents and other health and safety related incidents

Training courses on a variety of safety and hygiene topics are designed to provide Hanford Site-specific training for employees. These training courses are designed to fulfill certain initial and recurring training requirements specified in the Occupational Safety and Health topical area of Project Hanford Policies and Procedures. These training requirements are summarized in SNF Project implementing procedures.

17 4 1 3 Technical Reviews. Technical reviews verify that all changes are compatible with the facility design and that the proposed changes will not adversely affect facility safety, reliability, or operation. In addition to the reviews described above, SNF Project Engineering, as the design authority and the organization charged with safety and regulatory integration functions, is responsible for these technical reviews. The responsibilities and processes associated with the design authority function are provided in SNF Project administrative procedures.

17 4 1 4 ALARA Strategy. Minimization of exposure to workers and radiological safety is a fundamental precept of the SNF Project. ALARA planning will drive process and equipment selection to optimize cost-benefit, personnel exposure, and radiological safety. The SNF Project...
ALARA policy and program implements, through documents and administrative procedures, requirements from a number of documents. These requirements documents include:

- 10 CFR 835, "Occupational Radiation Protection"
- NRC Regulatory Guide 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable
- HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward Additional NRC Requirements
- G-10 CFR 835/E2-Rev 0, Workplace Air Monitoring
- HSRCM-1 Hanford Site Radiological Control Manual

Additional information on the SNF Project ALARA policy and program is presented in Chapter 7 of HNF-PRO implementing procedures will meet the ALARA requirements of the S/RID.

In addition to radiological application, ALARA principles are applied to exposure of personnel to hazardous materials in the workplace. Details of the ALARA program applicable to hazardous materials are contained in Chapter 8. The ALARA process is provided in SNF Project administrative procedures.

17.4.15 Unreviewed Safety Questions: DOE will define the facility authorization basis at the time of approval of the facility FSAR. The USQ process will be used, following DOE approval of the FSAR for each operational SNF Project facility, for evaluating physical and procedural changes and proposed tests and experiments to determine the existence of a USQ. The USQ process is described in SNF Project administrative procedures. SNF Project implementing procedures will meet the USQ requirements of Section 18.5 in the S/RID (HNF-SD-SNF-RD-001).

The USQ evaluation process allows the SNF Project facility management to make physical and procedural changes and to conduct tests and experiments without prior regulator approval provided that the activity does not involve a change to the TSRs or involve a USQ. A proposed change, test, or experiment involves a USQ (1) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the SAR may be increased, (2) if a possibility for an accident or malfunction of a different type than any evaluated previously in the SAR may be created, or (3) if the margin of safety as defined in the bases of the TSRs could be reduced.
17 4 1 6 **Topical Plans**  Topical plans are used for activities that apply to the SNF Project. This section describes the topical plans and documents required for the successful execution of the SNF Project. Some of the plans already have been developed, others will be developed before release of the final SAR. All plans will be revised as required. The review and approval authority of these and other SNF Project documents is reflected in the SFO Document Review and Approval Matrix contained in HNF-3552. In each case, the SNF Project will submit to DOE-RL an SNF Project-approved document. The plans include the following:

- **The SNF Project Integrated Safety Management Plan (ISMP)** (Williams 1997) presents a comprehensive safety management plan that ensures protection of the public, workers, and the environment. The safety management approach in this ISMP implements the provisions of HNF-MP-003 and provides for a defense-in-depth safety culture in the context of *Defense Nuclear Facility Safety Board Recommendation 95-2*. The ISMP will be updated to incorporate all elements of the DOE Implementation Plan for Defense Nuclear Facility Safety Board 95-2 (DOE 1996). The updates will address the seven safety management guiding principles and the safety management functions of the FDH sitewide ISMS.

  The ISMP (Williams 1997) implements the SNF Project Regulatory Strategy (WHC-SD-SNF-SP-002) and defines the safety processes, requirements, and responsibilities to be applied to the SNF Project. The ISMP also describes the necessary programs and processes required to maintain the existing K Basins Authorization Basis and to develop the Authorization Basis for new SNF Project processes and facilities that ensures compliance with applicable environmental, safety, and health requirements.

- **The Technical Baseline Document (WHC-SD-SNF-SD-003)** is the foundation for SNF Project functions and requirements. All activities within each SNF Project facility are traceable to the technical baseline.

- **The SNF Systems Engineering Management Plan (WHC-SD-SNF-SEMP-001)** implements the provisions of HNF-MP-007, *Systems Engineering Management Plan*, and describes the process used to ensure that the technical requirements and basic design criteria of project subelements are clearly defined throughout design, acquisition, construction, and operation, including those imposed on all project participants and subcontractors.

- **The SNF Project Systems Configuration Management Plan (HNF-SD-SNF-CM-001)** describes the controls used for technical requirements and basic design criteria of project subelements throughout design, acquisition, construction, and operations. This plan implements the provisions of HNF-MP-013 *Configuration Management Plan*, and also addresses the administrative requirements imposed on SNF Project participants and subcontractors. Included are the systems to ensure that changes to the project baselines are reviewed and approved by appropriate levels of management according to established thresholds.
• The SNF Project Acquisition Plan describes the project's acquisition strategy and provides the policy, guidance, and request for services, facilities, and systems acquired by the SNF Project. The project acquisition plan incorporates all DOE Order 4700.1 requirements for the acquisition plan. This document helps the facilitator reconcile user requirements and funding allocations and prioritize efforts.

• The Public Involvement Plan describes how the SNF Project will address the public coordination and involvement required by NEPA, the Tri-Party Agreement (Ecology 1994), and DOE guidance.

• The SNF Startup Plan (HNF-SD-SNF-SUP-003) ensures that the completed facility and all installed systems meet the established safety basis and functions and requirements. Detailed test plans and/or specifications and procedures are prepared, approved, controlled, and maintained. These test plans or specifications and procedures address testing requirements for all plant systems, subsystems, and individual pieces of equipment. The test planning and scheduling coordinate development testing with design, and plant testing with plans for construction completion, turnover, and plant startup. Training requirements for the above activities also are covered.

• The Permitting Plan identifies all preconstruction, preoperation, and operation regulatory requirements to the extent known. A strategy to obtain all required permits will be developed consistent with SNF Project requirements. The appropriate documentation and schedules to comply with the requirements will be included.

• The SNF Project Interface Control Plan (WHC-SD-SNF-CM-003) addresses the physical interfaces and controls required between project subelements. An important aspect of maintaining configuration is effective interface control. The SNF Project will establish an interface control process coordinated by an interface control working group. The interface control working group will manage all internal interfaces and coordinate with the Site Interface Control Working Group for resolution of external interfaces.

• The Technology Acquisition Plan (WHC-SD-SNF-CM-004) describes how technology acquisition services and technology support services will be managed. Included are designated process technology application and laboratory testing equipment adaptation and verification testing, the waste form qualification model, and data development.

• The SNF Project NEPA Integration Plan includes the impacts and interfaces of NEPA compliance related to all Hanford Site SNF.

• The Integrated ES&H Management System Plan (HNF-MP-003) stipulates the ES&H requirements that will be met in each SNF Project facility. The intent is to
apply the appropriate DOE rules, orders, and/or NRC technical requirements to the
various subprojects. This will be done with DOE's concurrence. In addition, the
project will comply with the Tri-Party Agreement (Ecology 1994). A VPP is being
implemented as a best management practice. These requirements and best
management practices protect workers, the public, and the environment and will be
applied using a graded approach. The SNF Project's environmental and safety and
health programs include environmental protection, occupational safety, fire
protection, industrial hygiene, radiological control, nuclear safety emergency
preparedness, and radioactive and hazardous waste management.

The SNF Project independent oversight organizations monitor compliance with the
requirements during operation of the K Basins and during planning, construction,
startup, and operation of the CVDF, the CSB and other SNF Project facilities or
activities. These oversight organizations, as well as workers in the facilities, are
authorized to stop work if an imminent safety, health, or environmental hazard is
observed.

17417 Lessons Learned In response to the requirements of Section 1.9 in the SRID
(HNF-SD-SNF-RD-001), DOE Order 5480 19, DOE Order 5480 26, and DOE O 232 1A, each
SNF Project facility maintains a lessons-learned program that includes the identification,
documentation, validation, and dissemination of lessons learned from design and construction
activities, from in-house events, and from other DOE or commercial nuclear facilities. The
lessons-learned program information is used to improve the design, construction, and operation of
the facility day-to-day operations, and to identify good practices and potential hazards. For
the Plutonium Reclamation Facility, identifies a number of programmatic deficiencies involving
emergency preparedness and hazardous materials that will be corrected in SNF Project programs.
In-house events are evaluated by the operations organization to determine whether the event or
condition should be included in the training program for operations personnel. Immediate training
is scheduled if necessary.

SNF Project personnel are required to review lessons-learned information contained in
required reading files and to incorporate appropriate practices into their daily work habits. This
information is also used in the development of training curricula and in the modification of
training materials as appropriate. Personnel are encouraged to participate in the process by
providing feedback on information distributed and to identify information for potential inclusion in
the process. The lessons-learned program is the responsibility of all SNF Project managers and is
administered by the lessons-learned point-of-contact. The overall process is described in
SNF Project administrative procedures.

17418 Consultation with Native American Tribes and Nations Consistent with the policy
issued by the Secretary of Energy, the SNF Project is committed to a consultation relationship
with Native Americans having treaty rights on the Hanford Site (WHC-SD-SNF-SP-006). The
nations and/or tribes include the Yakama Indian Nation, the Confederated Tribes of the Umatilla
Nation, and the Nez Perce Tribe. In an attempt to keep Native Americans apprised of all

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significant actions being undertaken by the SNF Project, briefing meetings are held with representatives from each tribe and/or nation.

Documents and strategies containing key decisional information are submitted to the tribes for review and comment concurrent with the decision making process at DOE-RL. The intent of this activity is to ensure that appropriate dialogue is held with the Native Americans and that their concerns are adequately resolved during the decision making process as the SNF Project proceeds towards its conclusion. WHC-SD-SNF-SP-006, *Tribal and Stakeholder Communication and Participation Strategy for the Spent Nuclear Fuel Project*, provides details of this process.

**17 4 1 9 Compliance with Applicable Federal, State, and Local Environmental Laws and Regulations** The SNF Project complies with a number of federal, state, and local environmental protection laws and regulations. The U.S. Environmental Protection Agency is the lead environmental regulatory agency on the Hanford Site for ensuring compliance with requirements promulgated pursuant to the *Comprehensive Environmental Response Compensation and Liability Act (CERCLA)* of 1980. The Washington State Department of Ecology is the authorized regulatory agency for compliance with the *Resource Conservation and Recovery Act of 1976* and WAC 173-303, "Dangerous Waste Regulations."

The applicable compliance requirements imposed by these agencies are documented as major, interim, and target milestones in the Tri-Party Agreement (Ecology 1994).

The SNF Project also regularly interacts with the Washington State Department of Health. The Department of Health has regulatory authority over radiological air emissions on the Hanford Site and must approve any new construction or operational activities undertaken by the SNF Project that may pose the potential for increasing radiological exposure to workers and the public. In addition, permitting authority for potable water supplies falls within the authority of the Department of Health.

Documentation related to compliance with applicable federal, state, and local environmental laws and regulations is processed in accordance with SNF Project administrative procedures.


SNF Project implementing procedures describe the process for complying with the *Price-Anderson Amendments Act of 1988* at the SNF Project facilities. Minor noncompliances are tracked using SNF Project administrative procedures. SNF Project administrative procedures also describe the process for trending of noncompliances to determine whether they are indicative.
of a programmatic breakdown. All of the above activities serve to enhance and protect the radiological health and safety of the public and workers at the SNF Project facilities.

17.4.2 Spent Nuclear Fuel Project Safety Management System Process

17.4.2.1 Integrated Safety Management  FDH has developed an ISMS plan, HNF-MP-003, which responds to Defense Nuclear Facility Safety Board Recommendation 95-2. This plan addresses implementation of an environmental management system consistent with the principles of ANSI/ISO 14001 and supports the Radiological Control Improvement Plan. The ISMS is based on the philosophies, principles, and requirements identified in DOE P 450.4 and the specification and guidance for environmental management systems (ANSI/ISO 14001). The objective of the ISMS is to provide a formal process to systematically integrate safety into work planning and execution processes to ensure work is performed safely to effectively protect the workers, public, and the environment.

Five essential safety functions or elements require consideration in the ISMS: (1) defining the scope of work, (2) identifying and analyzing hazards, (3) developing and implementing controls, (4) performing work within controls, and (5) providing feedback and continuous improvement. The following sections briefly discuss the management of each of the safety functions to ensure work is performed safely.

17.4.2.1.1 Define the Scope of Work  SNF Project work scope involves a variety of activities. These activities are performed in support of the SNF Project mission. To ensure work is performed safely, each activity is broken down into discrete work packages.

The proposed SNF Project ISMS identifies two work scopes: facility level and activity level. Facility-level work scope is directed and controlled based on the contractual agreement between the SNF Project and FDH. At the Hanford Site level, an integrated Hanford Site baseline is developed by FDH to integrate work scope and contractor interfaces between FDH, its subcontractors, and other Hanford Site contractors. Baseline authorization is the mutual acceptance and approval between FDH and DOE-RL for performing the work described by the integrated site baseline. DOE-RL approves the integrated site baseline and authorizes resources for the execution of work within a specific time period. FDH, in turn, authorizes work by its subcontractors through project authorization documents.

The SNF Project input into the integrated Hanford Site baseline comes in the form of a multi-year work plan that is reviewed by FDH and incorporated, if appropriate, in the final integrated Hanford Site baseline. The SNF Project prioritizes the work identified in the project authorization documents in a final multi-year work plan. The prioritization is based on mission compliance, cost, and risk containment objectives. At all working levels of the SNF Project, a risk-based prioritization process is used to establish a technically defensible logic for work planning and execution. This process is intended to balance priorities by using risk-based planning and resource allocation to meet regulatory requirements and control safety and environmental.

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hazards during the safe execution of work. Risk-based prioritization of work ensures that the most significant hazards are mitigated in the most cost-effective manner.

Activity-level work scope (e.g., work performed on SSCs within the SNF Project facilities) can be identified during facility walkdowns, during SSC surveillance and corrective and preventive maintenance, through engineered modifications to safety-related equipment, or through component or system failure. Work is prioritized according to the safety classification of the system and critical need to support mission requirements. Work at the SNF Project facilities is categorized into three categories based on the required skill of the worker and the hazards associated with the task to be performed. The three categories are (1) routine, (2) skill-of-the-craft, and (3) planning required.

17.4.2.1.2 Identify and Analyze Hazards Hazards are analyzed before initiating work activities. The hazard analyses are performed to identify conditions that could lead to undesired consequences to the public, the workers, and the environment. The hazards and accident analyses contained in Chapter 30 address the SNF Project facilities and their operations. The analyses contained in Chapter 30 are maintained and updated as part of the facility-level work scope. For new activities, including one-time and long-term activities, hazard analyses are performed based on the scope of the work activity and if appropriate are incorporated in the final SAR as a modification. The hazard analyses are designed to address all the hazards identified during the hazard identification activities including material at risk, energy sources, natural phenomena, external events, and common cause failures. The results of the analyses are used to support the identification of requirements and controls specifically related to the task analyzed (i.e., TSRs), training requirements, and the need for safety support (e.g., radiological, industrial hygiene, and safety fire protection).

17.4.2.1.3 Develop and Implement Controls The controls developed for the safe operation of the SNF Project are graded or tailored based on the results of the hazard and accident analyses described in Chapter 30 and on environmental requirements and analyses. The developed controls include TSRs, institutional controls and safety programs, facility-level operations documents, and environmental regulations. The SNF Project also prepares and maintains environmental documentation based upon the risks and hazards associated with the work being performed.

The TSRs, as discussed in DOE Order 5480.22 and Chapter 5.0, define acceptable conditions, safe boundaries, and administrative controls (or management) to ensure safe operations and reduce the potential risk to the public and onsite workers from uncontrolled releases of radioactive or other hazardous material or from radiation exposures caused by inadvertent criticality. The TSR controls are based on the hazard and accident analyses described in Chapter 30 and are considered necessary and sufficient for public safety, significant defense-in-depth, significant worker safety, and for maintaining potential consequences below risk guidelines. These controls define the safety envelope approved by DOE using safety analyses.

Integral to the implementation of the controls, the SNF Project maintains an auditable safety analysis program to maintain the controls. The auditable safety analysis program maintains...
the existing boundary for safe operation (establishing a safety envelope), as defined in Chapter 3 and the TSRs, and supports the identification, analysis, and control of previously unidentified hazards (e.g., hazards based on new work scope). The safety analysis program is used to ensure that each activity and operation within the facility or project is authorized under an approved authorization basis. An authorization basis is defined in DOE Order 5480.21 as those aspects of the facility design basis and operational requirements relied upon by DOE to authorize operation. The authorization basis management system (see Section 17.4.3.2) provides a process for developing authorization basis documents, evaluating and controlling changes, and implementing the authorization basis. All work, either in the form of proposed activities or new information (also known as a discovery), is screened against the authorization basis to verify that the proposed change or discovery is within the authorization basis.

Applicable standards and requirements are identified and documented in the S/RID (HNF-SD-SNF-RD-001). The S/RID defines the requirements agreed to by the contractor and DOE and is implemented through contractor policies and procedures. The S/RID captures the applicable safety requirements of the Code of Federal Regulations, DOE orders and regulations, Washington Administrative Code, and other requirement source documents (e.g., industry standards) that embody the necessary and sufficient requirements to ensure the safety of the public, the workers, and the environment. The process for identifying the applicable requirements is based on the results of the hazard and accident analyses described in Chapter 3. That is, the identified hazards, assumptions, and controls are compared to the entire list of requirements. Only those requirements applicable to the hazards, assumptions, and controls are listed.

Institutional controls, and safety programs are developed for the SNF Project based on the results of the hazard and accident analyses, the standards and requirements identification process, and the TSRs. These controls and programs, discussed in Chapters 6 through 17, address the prevention of inadvertent criticality, radiation protection, including ALARA, hazardous material protection, radioactive and hazardous waste management, quality assurance, emergency preparedness, initial testing, in-service surveillance, and maintenance, operational safety, procedures and training, human factors, provisions for decontamination and decommissioning, configuration control, and authorization basis management.


17.4.2.1.4 Perform Work Within Controls Safe work is conducted in accordance with the controls, procedures, and commitments developed from and contained in the authorization basis. At the activity level, the formality, degree to which work is proceduralized, and the degree of direct supervision is established based on the type and magnitude of the hazards. The confidence that the hazards are well known, the confidence in the hazard controls that are selected, the complexity of the work, the comprehensiveness of the safety envelope, and worker qualifications. Worker involvement plays an important role throughout the work planning process and the performance of work.
The SNF Project work planning process is used to prepare for work. This process defines work performance from planning to execution and includes confirming readiness, worker involvement, worker training, and worker competence, designating responsibilities and approval authority, and performing the work. During the planning process, workers and support personnel (e.g., radiological control personnel) provide input to work documents, perform site walkdowns, and review completed work documents. Through a combination of training, required reading, prejob planning meetings, and prejob safety briefings, workers are made cognizant of hazards and controls required to perform the work.

The SNF Project uses job hazard analyses to identify specific hazards and develop procedures, safety plans, instructions, or work permits to adequately address safety and health at the activity level. The safety instructions and controls identified following the job hazard analysis are written into the work package. Work packages contain numerous elements that are fundamental to worker safety. Within the work package, the hazards are identified and safety boundaries are established for the work to be performed. The applicable requirements are identified, worker skill needs are identified, and job site prerequisite conditions are identified.

SNF Project workers always have the power to "stop work" if any conditions exist that are felt to be unsafe or hazardous to the environment. Workers will not proceed and will halt work if instructions are unclear or inadequate or the controls are not adequate to control the hazards present.

17 4 2 1 5 Provide Feedback. Self-assessments are performed at SNF Projects to provide measurable feedback and support continuous improvement. The self-assessments are incorporated into work processes through the use of performance indicators, in-process monitoring, occurrence reports, management assessments, worker assessments, self-assessments, independent assessments, and the lessons-learned program.

The performance indicator programs at SNF Projects are generated on the basis of requirements, good management practices, and requests from management with conformance to DOE Order 5480 26.

The SNF Project procedure for occurrence reporting and reporting of operations information (Section 17 4 4) establishes and implements criteria for the identification, categorization, and notification of occurrences.

The management assessment program (Chapter 14 0) at the SNF Project is designed to integrate assessment schedules and use the resulting data for effective trending analysis. Program implementation includes guidance for preparation and performance of assessments. Assessment results are entered into the identified corrective action tracking system.

The SNF Project lessons-learned program (Section 17 4 1 7) encourages improvements in safety and operations by communicating Hanford Site and related industry-wide operating experience. Lessons learned are obtainable from a number of inputs including (1) Occurrence Reporting and Processing System final reports, (2) DOE Operational Experience Weekly,
(3) DOE Lessons Learned Information System, (4) Sitewide Lessons Learned Evaluation Request, 
(5) input from SNF Project personnel, (6) Health Physics Radiological Problem reports, 
(7) facility monitoring program, (8) DOE facility representative, (9) DOE Safety Notices, 
(10) results of assessments, (11) performance indicators and trends (12) post-job reviews, 
(13) Emergency Preparedness Drill program, (14) outage critiques, and (15) ORR Lessons 
Learned reports  Guidance is provided on the preparation and dissemination of lessons-learned 
reports, evaluation of action notices, and performing trend analysis

17 4 2 2 Integrated Environment, Safety, and Health Management System Roles and Responsibilities  FDH is ultimately responsible for protecting the public, environment, and 
workers from hazards associated with Hanford Site operations  To support safe and compliant 
work practices and to ensure an integrated approach to safety, the FDH ES&H and Performance 
Assurance organizations develop and communicate sitewide ES&H and Quality Assurance 
programs and policies to the major subcontractors for incorporation into their projects and 
activities

The SNF Project is responsible for establishing mechanisms that delineate responsibilities, 
establish work priorities, and use risk-informed planning to promote effective use of resources. 
The SNF Project also is responsible for designing and documenting work strategies (including 
stop-work authority) that effectively establish mitigative or preventive controls and consider 
hazard severity, worker qualifications, and operational conditions

The SNF Project has the responsibility to establish and maintain hazard analysis processes 
and capabilities and to use these capabilities to perform hazards analysis at all levels of the 
organization. The SNF Project has the responsibility to identify, establish, and implement hazard 
controls at all levels of its organization. Whether work is performed by the SNF Project or by its 
subcontractors, the SNF Project has the responsibility and accountability to perform work safely. The SNF Project is responsible for ensuring that work is proceeding safely and in concert with 
requirements. This obligation can be fulfilled in several ways but, as a minimum consists of a 
combination of three factors

- An established self-assessment program
- Monitoring against the measures established during the agreement process, including 
data collection and evaluation, reporting, and improvement actions taken
- Conducting assessments, routine observation, and awareness

The SNF Project is responsible for the implementation of ES&H requirements at SNF 
Project facilities and for performing work at the facility and activity levels safely. Line managers 
are responsible for the safety of workers and for the protection of the public and the environment. Line management accountability for planning and implementing ES&H requirements is 
fundamental to the management of the PHMC, and line manager responsibility is the key to the 
successful communication and implementation of an ISMS. Line managers establish the operating
procedures that clearly communicate the ES&H policies and directives, and line managers and supervisors also ensure employee understanding through the involvement of their employees.

At the worker level, employees are encouraged and, in some cases, required to participate in the completion of the enhanced work planning, job walkdowns, and monthly zero-accident council meetings. PHMC management requires that all employees participate in monthly safety meetings, whether it involves a local employee zero-accident council or other organized meeting devoted to a timely safety topic or discussion of the safety issues surrounding work. Employee input is welcomed for the purpose of continually improving the ES&H conditions at every work site.

Although DOE does not actively participate in the activities performed at the SNF Project, DOE is an active member of the ISMS process at the SNF Project through Hanford Site planning and oversight activities. Cognizant DOE employees, including workers, have the authority to stop work when they identify violations that significantly diminish the safety of workers, the public, or the environment. DOE performs the following activities:

- Defining mission and program objectives and providing direction and guidance to contractors
- Establishing contract and other agreements with the contractor that define the scope of the work and providing sufficient resources for the safe conduct of the work
- Establishing agreements with the contractor for safety expectations and standards appropriate for the work based on an understanding of the hazards
- Establishing its expectations for the rigor of the contractor's hazard analyses and establishing its role in review and approval of the hazard analyses and controls (e.g., TSRs for nuclear facilities) that are to be incorporated into the authorization basis
- Retaining sufficient oversight to determine whether applicable regulations for industrial hazards are being followed and, if they are not, what additional DOE action is required (through oversight DOE provides reasonable assurance that established controls conform to applicable regulations)
- Using readiness review, direct observation of contractor operations, and monitoring for significant events, and being aware of and monitoring the conditions under which work is performed
- Evaluating both DOE and PHMC performance using DOE assessments, assessing reported data, and performance measures, occurrence reports, and other data
- Providing feedback to support the continuous improvement process
An integrated management system has been developed and implemented to establish and maintain the SNF Project authorization basis (Section 17.4.3.2). This management system comprises four subsystems that are described in more detail in the following sections.

17.4.3 Configuration Control

In accordance with the requirements of Section 3.0 in the S/RID (HNF-SD-SNF-RD-001) and DOE-STD-1073-93, configuration management activities will establish and maintain consistency among design requirements, physical configuration, and facility documentation based on a graded approach. Types of equipment to be included in the configuration management program is identified based on the function provided by the SSCs and includes SSCs with safety design requirements (those necessary to protect offsite personnel, onsite personnel, and facility workers from nuclear and other hazards), environmental design requirements (those necessary to protect the environment from significant damage or to satisfy environmental requirements or permits), and mission design requirements (those necessary to avoid substantial interruptions of the programmatic mission or severe cost impacts).

The physical configuration conforms to the design requirements, which are established by the design output documents. The facility documentation accurately reflects the physical configuration and the design requirements. The facility documentation that supports facility operations includes the as-built configuration documents and facility procedures for activities. Changes to the facility design requirements are reflected in both the facility physical configuration and facility documentation. Changes to the facility physical configuration or its documentation must be supported by design requirements. The configuration control process is the responsibility of SNF Project engineering. Additional details of the configuration management program are contained in HNF-MP-013, HNF-SD-SNF-CM-001, *Spent Nuclear Fuel Project Configuration Management Plan*, and SNF Project implementing procedures.

In accordance with DOE-STD-1073-93, during planning for the configuration management program, initial assessments will be conducted to determine the strengths and weaknesses of existing programs and procedures with regard to determining where upgrade actions and resource investments are necessary. After the configuration management program upgrades are implemented, a horizontal slice assessment will be performed for each configuration management program element to determine if that element addresses identified weaknesses and is effective in accomplishing the configuration management functions.

Physical configuration assessments, or walkdowns, will be performed for representative sample SSCs to determine the degree of agreement between the physical configuration and the configuration depicted in the facility documentation. Physical walkdowns are included as part of the programmatic assessments conducted during the initial assessments and post-implementation assessments. If substantive discrepancies are discovered, appropriate immediate corrective actions will be developed to establish agreement between the physical configuration and the documentation. The corrective actions will include additional walkdowns to characterize the
problem and determine the extent of the problem. They will also include technical evaluations to
determine whether the physical configuration or the documentation needs to be changed.

SSCs within the configuration management program will be tested periodically to determine
if they are still capable of meeting their design requirements. This monitoring will also address
surveillance actions, periodic in-service inspections and tests, and other monitoring of SSCs to
ensure safe and reliable operation of the facility. Monitoring will also include measurements and
trending of data related to the actual aging degradation of equipment.

SSCs within the configuration management program will be tested after modification (and
before being turned over for service) to determine if it is capable of meeting its design
requirements. If a changed SSC fails to meet its post-implementation acceptance criteria,
turnover for operation will be postponed until either a technical review has been completed and
any follow-up actions are completed or until the SSC is returned to its original condition and
tested satisfactorily.

17 4 3 1 Configuration Control Process The configuration control process contains the
following five elements (DOE-STD-1073-93):

- **Identification** — Potential changes are identified and initiated by any project
  participant who determines that a departure from the existing technical baselines is
  warranted.

- **Evaluation** — After identification, the project participant who identifies the change
  performs an evaluation. This evaluation establishes that the potential change is both
  necessary and desirable. The precise evaluation process is determined by the
  importance of the change.

- **Approval** — Each change must be approved by the approval designator on the
  change documentation before authorization of work. Overall responsibility for
  approval resides with SNF Project facility engineering in the areas of design,
  configuration, and safety.

- **Authorization** — When a change is approved, it is processed through the required
  contract and work authorization cycles.

- **Implementation** — After the change has been approved and authorized, the changes
  are implemented and the required work performed.

Requirements associated with the development, review and approval release and
incorporation of changes to SNF Project engineering documents are implemented through the use
of the Engineering Change Notice as described in SNF Project implementing procedures. In
addition, HNF-SD-SNF-CM-001 states that all SNF Project related documents (functions and
requirements, analysis, design calculations, drawing specifications, procedures, etc.) containing
design-related information shall be developed, reviewed, and released in the document control.
system by use of an Engineering Data Transmittal. DOE-STD-1073-93 requires that only currently-approved revisions of documents under the control of the configuration management program should be used. See Chapter 12 of for further descriptions of the process for identifying the approved, current revision of a procedure.

Revisions to documents within the configuration management program should be completed in a timely manner. For each document within the configuration management program, the following information should be readily available: revision level, current status, document owner, information regarding pending changes, and other data necessary for control and tracking such as storage location and outstanding document change notices. See also Section 14.5 which states that following approval, controlled documents are released to specified users to ensure the latest approved revisions are available to personnel for use at the location where the work is being performed. See also Section 14.6.1.

Information Resource Management develops and maintains the project files and ensures that information is available to support the SNF Project facilities and that the information product is complete and accurate for the staging and interim storage of SNF in accordance with DOE-STD-1073-93. Information resources are managed throughout the information life cycle, including information creation, collection, processing, distribution, management, disposition, and retirement. Management of life cycle activities shall be targeted toward making information useful, available, and effective.

Creation of data files is consistent with the following requirements:

- For electronic information systems that produce, use, or store data files, disposition instruction for the data shall be incorporated into the system design.
- Up-to-date technical documentation is available for each electronic information system that produces, uses, or stores data files.

Permanent records are preserved, but records no longer of current use are promptly disposed of or retired. A records management program will be developed consistent with the requirements of federal law, Code of Federal Regulations, DOE directives, DOE guidelines, and DOE-established or accepted referenced standards. All records management program requirements are kept current and available for review.

Project files are developed and maintained in accordance with HNF-SD-SNF-CM-001, HNF-SD-SNF-MP-001, Spent Nuclear Fuel Project Data Management Plan, and SNF Project implementing procedures.

17 4 3 2 Spent Nuclear Fuel Project Safety Management System

The SNF Project safety management system has been developed and implemented to establish and maintain the SNF Project facility authorization basis. This safety management system comprises four subsystems: the technical baseline management system (Section 17 4 3 2 1), the risk management system (Section 17 4 3 2 2), the authorization basis management system (Section 17 4 3 2 3), and...
the safety management programs system (Section 17 4 3 2 4) These subsystems operate as an integrated package, sharing information and resources

17 4 3 2 1 Technical Baseline Management System A total baseline will be established for all SNF Project facility activities to the completion of the project. The technical baseline, discussed below, is the basis for the schedule and cost baselines that are reflected in HNF-SP-1104, Spent Nuclear Fuel Project Multi-Year Program Plan

In addition to the technical baseline, other baselines include Schedule Baseline Control, Cost Baseline Control, Fuels Management, Project Work Authorization, Project Performance and Control, Project Startup, Management Reporting, and Management Reviews, and Management Planning. Details of these may be found in WHC-SD-W441-PMP-001, Project W-441 Cold Vacuum Drying Project Management Plan, and WHC-SD-W379-PMP-001, Project W-379 Canister Storage Building Project Management Plan

Technical Baseline and Control The technical baseline management system is used for management and maintenance of facility documents related to (1) configuration of SNF Project facility SSCs, (2) facility site description and characteristics, (3) inventory of hazardous materials in the facility, (4) facility requirements, and (5) safety basis documents that support the facility authorization basis (WHC-SD-SNF-SD-005)

The technical basis for the SNF Project facilities is the approved technical baseline. It consists of the following documents

- Project Management Plan (HNF-3552, WHC-SD-W441-PMP-001, and WHC-SD-W379-PMP-001)
- Systems Engineering Baseline Document (WHC-SD-SNF-SEMP-001) for the scope defined in the Project Management Plan
- Process strategy document, a product of system engineering methodology, Independent Technical Assessment Group evaluations, and other studies
- Work Breakdown Structure, Work Breakdown Structure Dictionaries and Index (included in the multi-year program plan [HNF-SP-1104])
- End point criteria, facility-specific agreements between the current caretaker of a facility and the future caretaker of the facility regarding the configuration and condition of the facility at the point of turnover
- Acceptance criteria and memoranda of agreement or understanding, agreements between the SNF Project facility and organizations outside of the SNF Project, regarding the transfer of material from one organization to the other or other agreements or understandings regarding the split of responsibilities
Functions and requirements documentation for the SNF Project facilities

The SNF Project facility is responsible for the overall development, implementation, and maintenance of the baseline documentation. The SNF Project facility ensures that configuration management activities and systems engineering activities are performed, while better defining and controlling the project baseline and associated documentation. These activities are applied to all systems and subsystems necessary to achieve all functional requirements and deliver all products to satisfy the integrated technical baseline and overall project objectives.

At all times during the life of the SNF Project facility, current configuration will be maintained in orderly and auditable project files. These project records and files include, but are not limited to system descriptions, system specifications, conceptual and definitive system designs, system and material inspection reports, test reports, operating and surveillance procedures, and vendor documentation.

Structures, Systems, and Components Management

SSC configuration is defined in terms of engineering documents (e.g., drawings, design analyses, design requirements and criteria, equipment lists, vendor information). The SNF Project facility SSCs management subsystem (1) identifies SSCs important to safety, (2) maintains the documents that describe the SSC configuration, (3) provides information about the status of items under configuration control, and (4) provides a process for evaluating and implementing proposed changes. Information about SSC configuration changes also is incorporated into the following subsystems: risk identification, establish prevention/mitigation, authorization basis control, and site safety management.

Site Description Management

Site description information includes meteorological, seismic, lightning, flood, volcanic, and geophysical data. The site description management subsystem entails a process for identifying data important to the safety management system. The site description management subsystem is used to specify appropriate media for the documentation and retrieval of the data and the processes to control changes to the site description data. Changes in the site description also are incorporated into the following subsystems: risk identification, authorization basis control, and site safety management.

Hazardous Material Inventory Tracking

The hazardous material inventory tracking subsystem entails a process for identifying and documenting important attributes (type, quantity form, process or storage condition, and location) of the hazardous materials associated with the SNF Project facility. This hazardous materials information is used to determine the facility hazard categories and resulting authorization basis documentation requirements. Inventory tracking also ensures that any changes to hazards will be identified and evaluated. Changes in hazardous materials information are incorporated into the risk identification and authorization basis control subsystems.

Requirements Management

The requirements management subsystem entails a documented process for identifying and allocating the requirements and systems engineering functions and requirements important to the SNF Project facility safety management system. Changes in requirements are assessed and incorporated into affected subsystems.
HNF-SD-SNF-RD-001 will be used with the systems engineering functions and requirements for the SNF Project safety management system

17 4 3 2 2 Risk Management System The SNF Project facilities will adopt a methodology to minimize the overall risk through timely, documented and defensible project decisions consistent with HNF-MP-005, Risk Management Plan. All major decisions, as documented in a decision log, will be tied to the project critical path and will include an assessment (qualitative or quantitative) of the project risk associated with each proposed alternative solution. Project risk assessments will consider the cost, schedule, and technology risk associated with the proposed alternative solutions and will require decisions in a timely manner to support the project critical path schedule, thus minimizing the overall risk to downstream project activities.

As a guiding principle, and to minimize overall project risk the SNF Project facilities will favor low-cost, commercially available, or tried and proven design solutions over technically unproven options requiring research and development, specialty design, and fabrication. This is consistent with guidance in WHC-SD-SNF-SD-003, Spent Nuclear Fuel Project Technical Baseline Document.

Risk Identification Risk identification entails identification of accident initiators and establishes the processes for performing hazard and accident analyses to determine the frequency and consequences associated with various postulated events. Actual or proposed changes in hazard identification (e.g., type, quantity, form, process or storage condition and location) are incorporated into the hazardous material inventory tracking system in the site characteristics documentation.

Changes in frequency or consequence are considered when performing the following activities: risk evaluation, establish prevention/mitigation, and authorization basis control. Risks originating outside the scope of the SNF Project facilities that may affect the functioning of facility safety-class or safety-significant SSCs, or that may initiate accidents within an SNF Project facility, will be similarly considered.

Risk Evaluation Risk evaluation includes a process for evaluating identified risks against the authorized safety basis to determine whether any controls are necessary or prudent to reduce the overall risk. Changes in frequency or consequences of the hazard analyses (or accident analyses) or changes in the approved risk guidelines are to be factored in when performing this activity. Changes in the level of risk also are to be factored in when establishing prevention/mitigation and when performing the authorization basis control activities.

Establish Prevention or Mitigation Establishing prevention or mitigation includes a process for determining the appropriate SSCs and administrative controls required to reduce identified risks. This process makes use of the risk guidelines and implements appropriate defense in depth. Changes in the level of risk associated with identified accidents and changes in the SSC administrative controls credited in the analyses are factored in when performing this activity.
Changes in the list of required SSCs and administrative controls are inputs to the SNF Project facility SSC management and the authorization basis control activities

17 4 3 2 3 Authorization Basis Management System The authorization basis management system provides a process for developing authorization basis documents, evaluating and controlling changes, and implementing the authorization basis.

Authorization Basis Document Development The authorization basis document development activities provide a process for developing and revising authorization basis documents and for maintaining a list of authorization basis documents. Changes to the authorization basis documents are factored into the authorization basis control and authorization basis implementation activities.

Authorization Basis Control Authorization basis control activities include a process for screening, evaluating, documenting, tracking, and resolving authorization basis changes. The USQ process is a key component of this activity. Actual and proposed changes in the facility SSCs, site description, facility hazards, facility risks, safety basis documents, authorization basis documents, and operating and maintenance procedures must be considered when performing this activity. Discoveries and occurrences must also be factored into authorization basis control activities. Changes to the authorization basis are considered when performing authorization basis implementation, authorization basis document development, and safety basis document management activities.

Authorization Basis Implementation Authorization basis implementation includes a process for implementing the authorization basis in the appropriate documents and procedures and developing the requisite training. Changes in the authorization basis documents and in the functional classification of SSCs are factored into authorization basis implementation activities. Authorization basis implementation activities include tracking the implementation documents and producing a list of documents of use for authorization basis control and safety basis document management activities.

The process of developing the authorization basis includes the following activities:

- Definition of a Requirements Basis The requirements source documents (e.g., codes and standards) establish the requirements for design, safety analysis, environmental permitting, construction, and operation. The requirements basis for the SNF Project facility is the SNF Project technical baseline WHC-SD-SNF-SD-005, Spent Nuclear Fuel Project Technical Baseline Description, and HNF-SD-SNF-RD-001.

- Establishment of Functions and Requirements Using the requirements bases, systems engineering analyses identify the functions to be performed and the specific requirements to be met.

- Develop Design and Safety Analysis Documents The functions and requirements establish the scope and requirements to be met by the design and safety analysis.
documents. The functions and requirements are typically imposed by the performance specifications. The result of these activities is the creation of the detailed design documentation, SAR, and TSRs (or amendments) that support construction.

- Development of Operations Control Documentation. Authorization of operation is based on a comprehensive review (ORR and readiness assessment, as appropriate) to ensure the documentation, systems, and support functions are in place. Key documents supporting operations are the SAR and TSRs (or amendments as needed), the S/RID (or updates, as necessary), and operating procedures that implement the safety basis. The SAR and TSRs are developed during design. The final S/RID (or updates) are prepared based on all of the above activities. The operating procedures are developed during construction, and personnel training is completed during startup testing. SNF Project facility management is committed to the development and effective implementation of operating, technical, and administrative procedures. The process for developing, maintaining, distributing and training on new and/or modified procedures is described in Chapter 12.

Environmental permits are a portion of the authorization basis and are obtained as necessary to support design, construction, and operation activities in accordance with the ISMP (Williams 1997). The process for securing the necessary environmental permits is described in SNF Project administrative procedures.

Actions to be taken by the SNF Project facility organizations to implement new or revised safety bases documents are provided in SNF Project administrative procedures.

Systems Engineering. The SNF Project has several critical interfaces among its facilities, organizations, and suppliers. To ensure that these interfaces are compatible, the SNF Project implements systems engineering as an extension of good engineering and management practices (WHC-SD-SNF-SEMP-001) and HNF-MP-007. The implementation of systems engineering on the SNF Project is the sole responsibility of the SNF Project. A review will be performed to ensure that overall project objectives of cost, schedule, and technical performance are achieved during design of systems and subsystems.

Implementation of systems engineering on the SNF Project is to achieve:

- Clearly identified systems
- Clearly identified management authority
- Clearly identified design authority
- Clearly identified decision methodology
- Timely decision making
- Documented and maintained baseline
- Demonstration of product performance

Additional detail on the above SNF Project commitments is contained in HNF-3552.
SNF Project requirements are managed as a part of the systems engineering process. Requirements include the following:

- Establish and manage the requirements basis
- Establish the functions to be performed and allocate the requirements associated with those functions
- Develop design and safety analysis for equipment and systems that perform the functions and implement the requirements
- Construct equipment and implementing systems in accordance with the designs and requirements
- Develop or update S/RIDs
- Operate the equipment and systems within design and safety limits and requirements and in accordance with TSR and/or operating safety requirements, in accordance with approved operating procedures

Specific responsibilities related to configuration management of the S/RID program are presented in SNF Project administrative procedures.

17 4 3 2 4 Site Safety Management Programs System Numerous other administrative management programs interface with the safety management system. Examples of these administrative management programs include prevention of inadvertent criticality, radiation protection, hazardous material protection, radioactive and hazardous waste management, quality assurance, emergency preparedness, initial testing, in-service surveillance, and maintenance, operational safety, procedures and training, human factors, provisions for decontamination and decommissioning, and management, organization, and institutional safety provisions. Each program is evaluated to determine its impact on the safety management system and to determine when changes to that program must be factored into the SNF Project facility safety management system.

17 4 4 Occurrence Reporting

Section 1 6 of the S/RID (HNF-SD-SNF-RD-001), and DOE O 232 1A (as supplemented by DOE M 232 1-1A) establish requirements for reporting and processing operations information related to emergencies, unusual occurrences, and off-normal occurrences. This system provides equivalency with the evaluation process and reporting schedule requirements of Title 10, Code of Federal Regulations, Part 50 "Domestic Licensing of Production and Utilization Facilities," Section 50 55(e), "Conditions of Construction Permits" (10 CFR 50), for identification of defects and failure to comply associated with substantial safety hazards. Equivalency with the contractor evaluation and reporting requirements of Title 10, Code of Federal Regulations Part 21.
"Reporting of Defects and Noncompliance" (10 CFR 21), for potential defects and noncompliances also is achieved with the occurrence reporting system. Vendors report defects to the SNF Project for reporting to DOE in a manner equivalent to reporting to the NRC under 10 CFR 21.

The SNF Project facility manager, or a designee, has ultimate responsibility for the requirements of DOE O 232 1A. Once the appropriate response is initiated by personnel to stabilize or return the facility to a safe condition, the event or condition is categorized and the Occurrence Notification Center duty officer for unusual occurrences or emergencies and the DOE-RL facility representative are notified. The duty officer assigns a reportable occurrence number and makes other required notifications. During the entire process of notification and reporting, the facility representative and duty officer use the current management chain established for the line organization in providing program direction to the SNF Project.

SNF Project administrative procedures provide the process associated with the investigation, timely reporting, trending, and processing of occurrence reports concerning events or conditions that could have adverse implications for safety, health, environment, quality assurance, or security in accordance with the requirements of DOE Order 5482 1B, Environment Safety and Health Appraisal Program, and DOE Order 5484 1. SNF Project administrative procedures identify the responsibilities and process for identifying administrative and technical deficiencies. They also identify the process for ensuring deficiencies are identified and disseminated to the appropriate tracking and trending systems as required by 10 CFR 820.

17 4 4 1 Notification Procedures As required by 10 CFR 835, Title 40, Code of Federal Regulations, Part 355, “Emergency Planning and Notification” (40 CFR 355), and DOE O 232 1A, events that require notification of DOE personnel (and state and local officials, when appropriate) require a systematic, controlled method for gathering and transferring information. Procedures have been developed to support the fulfillment of DOE requirements.

Management ensures that all appropriate personnel are notified of emergencies when required. Names of primary and alternate contacts and current phone and pager numbers are readily available to the individual who is assigned to make the notifications. All notifications are documented in the manager's or supervisor's narrative log. The procedures for emergency notifications are discussed in Chapter 15 0.

17 4 4 2 Corrective Action Determination The actual or probable causes of a problem are evaluated by one or more techniques or methodologies to establish a final root cause. As required by DOE Order 4330 4B, an acceptable root cause meets three criteria: (1) its correction will prevent recurrence of the unplanned occurrence, (2) its correction will be feasible, and (3) its correction will not adversely impact safety, reliability, or operational goals.

Appropriate corrective action is established for each event investigation, and specific personnel are assigned responsibilities for the corrective action as specified by DOE Order 5480 19. Corrective action can take the form of procedural changes, training, design, etc.
modifications, or changes to administrative controls. The final approval for corrective action is made by the SNF Project facility manager.

An investigative report is prepared in accordance with DOE Order 5480.19 that contains a description of the event, a discussion of the impact of the event root cause, lessons learned, and the proposed corrective actions. The investigative report is approved by the SNF Project facility manager and reviewed by appropriate supervisors, managers, and the safety review committee.

17 4 4 3 Event Training In accordance with DOE Order 5480.19, in-house events are evaluated to determine whether the event should be included in the training program for appropriate SNF Project personnel. Sometimes, because of the severity or possible safety consequences of some events, immediate training is appropriate.

17 4 4 4 Feedback and Trending As described in the lessons-learned program occurrence reports will be used to feed back relevant operational history into the SAR. Trending of occurrences will result in safety analyses that reflect both current and past operating practices and will be used to verify continuous improvement in SNF Project facility management. Feedback trending, and root cause evaluation as part of the investigation process all serve to minimize recurrence.

Patterns of deficiencies, such as operator errors or inadequate procedures, are trended. A periodic summary report of events, causes, and trends is submitted to appropriate managers. Applicable material is incorporated into training programs.

17 4 4 5 Tracking As required by 10 CFR 820, tracking systems will be established to provide:

- Timely self-identification of nuclear safety deficiencies
- Prompt and complete reporting of such deficiencies to DOE
- Root cause analysis of nuclear safety deficiencies
- Prompt correction of nuclear safety deficiencies in a manner that precludes recurrence
- Identification of modifications in practices or facilities that can improve public or worker radiological health and safety

17 4 5 Operational Readiness Reviews

ORRs are required by Section 17 in the S/RID (HNF-SD-SNF-RD-001). ORRs provide a structured methodology for verifying that a facility is ready to be operated in a safe and controlled manner. The process will be applied to SNF Project facilities on a graded approach which addresses all key elements of the ORR, ranging from short, simple, low-level efforts to
comprehensive efforts covering all hardware, personnel, and procedures depending on the complexity and hazards. The ORR process will include a verification of the level of compliance with ES&H orders and Occupational Safety and Health Administration regulations.

DOE O 425.1 and DOE C 425.1 establish the actions to be taken and assign responsibilities for authorizing the startup of new facilities and restart of nuclear facilities. RLID 425.1 implements the requirements of DOE O 425.1 at the Hanford Site. ORRs will be required for startup or restart of the SNF Project facilities under the following circumstances:

- Initial startup
- Restart after an unplanned shutdown directed by DOE management for safety or other appropriate reasons
- Restart after an extended shutdown of 12 months
- Restart after substantial process, system, or SNF Project facility modifications that require changes in the safety basis previously approved by DOE
- Restart after a shutdown because of operations outside the safety basis
- When deemed necessary by DOE management.

SNF Project implementing procedures, in conjunction with DOE O 425.1, DOE C 425.1, RLID 425.1, and DOE-STD-3006-95 define the process by which operational readiness is determined. This includes the breadth of the ORR development of a plan-of-action, development of implementation plans, tracking and closure of findings, and certification and verification that the facility is ready to start up or restart.

17.4.6 Safety Culture

This section summarizes the policies and programs in place to facilitate evaluation to ensure that (1) safety awareness is a primary concern to management and (2) employees at all levels of the SNF Project are aware they have an obligation to ensure operations are conducted safely.

17.4.6.1 Fluor Daniel Hanford, Incorporated

FDH supports safe and compliant work practices through its integrated Environmental, Safety and Health Program. The heart of the FDH Environmental Safety and Health Program is the firm commitment by all managers, supervisors, and employees to prevent accidents and the conditions that could lead to injuries or endanger the environment. Through this emphasis on safety first, FDH management is improving the Hanford Site safety culture and demonstrating its commitment to affect change in the following key areas: management commitment, employee involvement, environmental safety and health training, worksite analysis, and hazard prevention and control.
17.4.6.2 Spent Nuclear Fuel Project  The fundamental principles at the core of the SNF Project management approach to safety are summarized below. These principles support the FDH safety-first emphasis and are infused throughout all layers of the organization, guiding day-to-day decision making and conduct.

- The development of safety culture is compliant with the VPP elements described in Section 17.4.1.2.

- The authorization basis establishes the bounds within which all operations may be safely conducted. This principle permeates every chapter of the SAR. The SNF Project has demonstrated its commitment to this principle by identifying and documenting the safety basis of every SNF Project-managed facility and activity and by implementing physical and administrative controls appropriate to risk in order to protect the public, the workers, and the environment against identified hazards.

- Management is responsible for providing leadership and support to site workers. This responsibility includes establishing goals and standards for work activities and providing the resources and materials necessary to allow workers to succeed. Management fulfills this responsibility through formal planning, coordinating, reporting, and budgeting processes, and through formal and informal interaction with workers.

- Work is performed in accordance with established controls. This ensures repeatable, predictable operation that complies with regulatory requirements and implements safe work practices. The rigorous approach to procedural development, the performance-based approach to training, and the emphasis on following procedures when conducting operations, demonstrate SNF Project commitment to working in accordance with established controls (see Chapters 11.0 and 12.0).

- Workers are responsible for ensuring excellence and are individually responsible for their own safety and the safety of their coworkers and the facility. As noted in Chapter 12.0, the concept of individual responsibility is exemplified by the fact that every worker has the authority to stop work if a procedural step cannot be implemented safely (see Chapters 7.0, 8.0, and 14.0). Stop-work authority is emphasized in the orientation training provided to all Hanford Site workers. Worker empowerment is further emphasized in SNF Project implementing procedures, which contain guidance that encourages and requires employees to immediately notify their supervisors upon observing any event or condition adverse to safety, health, quality safeguards and security, operations, or the environment. An employee concerns program provides another avenue for identifying problems to SNF Project management if an employee is dissatisfied with resolution through normal channels. Employees may also relay concerns directly to regulatory authorities if other alternatives do not result in correction of the problem.
Safety culture includes characteristics and attitudes in organizations and individuals that establish that, as an overriding priority, safety issues receive the attention warranted by their significance (IAEA 75-INSA-4). In addition to the above, the SNF Project promotes a company-wide safety culture through the following activities:

- Increasing individual awareness of the importance of safety, both on and off the job (i.e., through periodic safety meetings)
- Training personnel in safety skills, such as recognition and reporting of unsafe acts or conditions and conducting operations in a safe manner
- Achieving a safety commitment by all personnel
- Using the checks and balances of audit and review practices for meaningful, high-quality self-appraisals
- Developing systems and corrective measures that result in preventive rather than responsive actions
- Encouraging individual responsibility through formal assignment and description of duties
- Developing a high degree of personal accountability for safety in all employees, and acceptance of the philosophy that all injuries and accidents can and must be prevented

17.5 REFERENCES


Comprehensive Environmental Response Compensation and Liability Act (CERCLA) of 1980, 42 U S C 9601, et seq


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DOE M 232 1-1A, Occurrence Reporting and Processing of Operations Information U S Department of Energy, Washington, D C

DOE O 232 1A, Occurrence Reporting and Processing of Operations Information U S Department of Energy, Washington, D C

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HNF-MP-001, Management and Integration Plan, Fluor Daniel Hanford, Incorporated, Richland, Washington


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Figure 17-1  The Fluor Daniel Hanford, Incorporated Organization
Figure 17-2  Project Hanford Management Contract
Management and Integration Roles

Enterprise Companies

U.S. Department of Energy
Richland Operations Office

Fluor Daniel Hanford, Incorporated

River Corridor

Nuclear Material Stabilization

Waste Management

Spent Nuclear Fuel

Major Subcontractors

Provide/coordinate infrastructure and cross-cutting services
Establish operational standards for subcontractor minimums
Establish tactical measures
Integrate work/support processes and plans
Establish goals and evaluate effectiveness of subcontractors

Establish strategic performance measures
Define outcomes/deliverables
Establish minimum site standards

Coordinate regulatory/stakeholder initiatives
Define success criteria/values
Establish boundaries for integration

Provide U.S. Department of Energy
Spent Nuclear Fuel priorities
Coordinate regulatory/stakeholder initiatives

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Figure 17-3  Spent Nuclear Fuel Project Organization

Spent Nuclear Fuel Organization

- Vice President
- Project Director

- Technical/Administrative Staff
- Other Site SNF Disposition
- SNF Transfer - CVDF
- K East - K West Fuel Removal
- Sludge Debris Water Cleanup
- Construction Projects
- CSB SNF Waste Storage

- Engineering
- Quality Assurance
- SM & Emergency Planning
- Startup Integration
- Project Controls
- Perform Impl/Reg Ser
- Contracts
- Radiological Controls

CSB = Canister Storage Building
CVDF = Cold Vacuum Drying Facility
SNF = spent nuclear fuel
Figure 17-4  Canister Storage Building Spent Nuclear Fuel Waste Storage/K East–K West Fuel Removal/Spent Nuclear Fuel Transfer — Cold Vacuum Drying Facility Operations
Figure 17-5  Sludge, Debris, Water Cleanup and Other Site Spent Nuclear Fuel Disposition Operations
Figure 17-6  Startup Integration

Startup Integration

- Startup
- Operations Readiness Review
Figure 17-7 Maintenance
Figure 17-8  Radiological Control

Radiological Control

- Operations
- Support

- Relief
- Fuel Retrieval System/Integrated Water Treatment System Startup
- K West
- K East
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SAFETY STRUCTURES, SYSTEMS, AND COMPONENTS
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LIST OF TERMS

ASHRAE  American Society of Heating, Refrigerating, and Air-Conditioning Engineers
CVDF   Cold Vacuum Drying Facility
DBA    design basis accident
DBE    design basis earthquake
DOE    U S Department of Energy
DPIT   differential pressure-indicating transmitter
FSAR   final safety analysis report
HEPA   high-efficiency particulate air (filter)
HVAC   heating, ventilation, and air conditioning
MCO    multi-canister overpack
MCS    monitoring and control system
NRC    U S Nuclear Regulatory Commission
PES    process equipment skid
PLC    programmable logic controller
PWC    process water conditioning
SCHe   safety-class helium
SCIC   safety-class instrumentation and control
SNF    spent nuclear fuel
SSC    structure, system, and component
TSR    technical safety requirement
UBC    Uniform Building Code
VPS    vacuum purge system

8-4-4 eight-hour initial vacuum cycle, four-hour subsequent vacuum cycles, four-hour return to pressure between vacuum cycles

ISO & PURGE the MCO is isolated by closure of the VPS, general-service helium system, PWC system, and deionized water isolation valves and the SCHe system is actuated
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B4 0 SAFETY STRUCTURES, SYSTEMS, AND COMPONENTS

B4 1 INTRODUCTION

This chapter provides details on those structures, systems, and components (SSCs) that are necessary for the facility to meet offsite release limits and to satisfy onsite risk evaluation guidelines, provide significant defense in depth or contribute to worker safety. This chapter meets the requirements of DOE Order 5480.23, Nuclear Safety Analysis Reports, follows the format guidance of DOE-STD-3009-94, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports, and includes the following information:

- A list of the safety-class SSCs

  Subsections contain details of safety functions, system descriptions, functional requirements, system evaluations, and assumptions requiring technical safety requirements (TSRs) to ensure performance of the safety functions.

- A list of the safety-significant SSCs

  Subsections contain details of safety functions, system descriptions, functional requirements, system evaluations, and assumptions requiring TSRs to ensure performance of the safety functions.

There were no SSCs identified in Chapter B3 0 for defense in depth or worker safety that were upgraded to safety-significant.

In general, safety-class SSCs are those items required for protection of the offsite environment and the public. Safety-class SSCs include those items designated as safety class, in accordance with DOE Order 6430.1A, General Design Criteria. Safety-class SSCs also encompass items that are designated as “important to safety” and have been classified as Category A, as defined in Item 29 of HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward Additional NRC Requirements.

Safety-significant items are those SSCs required for the protection of onsite personnel not directly involved in facility operations. Safety-significant SSCs also encompass those items that have been designated important-to-safety Category B, in accordance with Item 29 of HNF-SD-SNF-DB-003. SSCs that would prevent or mitigate an onsite fatality or protect large numbers of facility workers (not industrial safety), or those SSCs that would prevent or mitigate toxic chemical exposures, also are designated safety significant.

All SSCs that are not classified as safety class or safety significant are general-service SSCs. Those SSCs that have been designated as important-to-safety Category C, in accordance with Item 29 of HNF-SD-SNF-DB-003, also are classified as the general-service SSCs. General-service SSCs protect workers from standard industrial hazards or are controlled by site safety programs.
The Cold Vacuum Drying Facility (CVDF) is classified as a hazard category 2 nuclear facility and has the potential for an accident resulting in significant onsite and offsite consequences. The CVDF has associated safety-significant SSCs to provide protection to the onsite workers, and as appropriate safety-class SSCs to provide protection to the offsite public. Because of their public protection function, safety-class SSCs generally require more formality in establishing functional requirements and performance criteria than safety-significant SSCs.

The accident analysis in Chapter B3.0 identifies six design basis accidents (DBAs) whose unmitigated consequences bound the accident sequences and scenarios identified in the hazard analysis process. Of these DBAs, two had safety-class consequences. The thermal runaway accident that could occur after draining has safety-class consequences, and the multi-canister overpack (MCO) overpressurization accident that could breach the MCO and blow down from high pressure also has safety-class consequences.

Safety-class systems and components are designed to prevent these accidents or mitigate the offsite consequences to within applicable limits. For the MCO overpressurization accident, redundant pressure management features discharge in a fashion that facility high-efficiency particulate air (HEPA) filtered ventilation systems mitigate the onsite consequence to within risk evaluation guidelines. Safety-class systems and components are designed to prevent these accidents from occurring during credible conditions. For specific instances when the onsite consequence exceeds the guidelines while the offsite consequence is within limits and the event scenario is not demonstrated incredible, the ventilation systems are credited with safety-significant mitigation of the onsite dose.

The other four DBAs (gaseous release, liquid release, MCO external hydrogen explosion, and MCO internal hydrogen explosion) have safety-significant unmitigated consequences. The gaseous and liquid release scenarios are mitigated by maintaining negative differential pressure within the facility, thereby confining radioactive releases within the facility using HEPA-filtered exhaust ventilation systems that are designated safety-significant. External hydrogen explosion release scenarios are prevented in the exhaust duct by hydrogen mitigation strategies incorporating restrictions in the vent paths and dilution flow from the ventilation system. The internal hydrogen explosion is prevented for credible scenarios by accomplishing MCO hydrogen management strategies using some of the same safety-class systems designed to prevent the thermal runaway and additional components of existing processing systems designated safety-significant. In scenarios where blow-back into the process bay is not incredible, the ventilation systems are credited with safety-significant mitigation of the onsite dose.

Each accident analysis section in Chapter B3.0 concludes with a summary of safety SSCs that provides the basis for this chapter. A summary listing of the accident categories and the designated safety SSCs that prevent these accidents or mitigate their consequences is provided in Table B4.1. Table B4.2 provides a summary list of the safety SSCs identified and provides the level of safety credited for each accident category. Many SSCs provide a safety function for prevention or mitigation of more than one accident. Detailed definitions of safety-class and safety-significant SSCs are provided in Section 3.4.1 of the Spent Nuclear Fuel (SNF) Project Final Safety Analysis Report (FSAR).
Table B4-1  Safety-Class and Safety-Significant Structures, Systems and Components Summary List  (16 sheets)

<table>
<thead>
<tr>
<th>Representative and bounding accident (Chapter B3 0)</th>
<th>Safety structures, systems and components*</th>
<th>Safety function</th>
<th>Summary justification of safety function *</th>
<th>Controls</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gaseous release (Section B3 4 2 1)</td>
<td>CVDF structural features</td>
<td>X</td>
<td>Provide confinement for gaseous release inside process bays or process water tank room</td>
<td>(Design feature Section B5 6 5)</td>
</tr>
<tr>
<td></td>
<td>Cask-MCO safety-class components</td>
<td>X</td>
<td>Provide pressure boundary for confinement of gaseous radioactive materials</td>
<td>(Design features Sections B5 6 3 and B5 6 4)</td>
</tr>
<tr>
<td></td>
<td>PWC system transfer line</td>
<td>X</td>
<td>Prevents pipe leak in process bays</td>
<td>(Design feature Section B5 6 7)</td>
</tr>
<tr>
<td></td>
<td>Process bay local exhaust HVAC and process vent system (HVACC/PV system)</td>
<td>X</td>
<td>Mitigates gaseous release into process bay through HEPA filtration before discharge to outside of the facility</td>
<td>Some airborne contamination in a process bay will be swept into the HVACC/PV system.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Provides hood isolation dampers</td>
<td>The restart function actuates dampers to reopen.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>With standby power interface, maintains a process bay DP when the HVAC system is not operable</td>
<td>Operation of local exhaust on standby power will ensure DP sufficient for confinement function during facility power outages, which enhances facility safety position.</td>
</tr>
<tr>
<td></td>
<td>Process general supply/exhaust HVAC system (HVACD system)</td>
<td>X</td>
<td>Mitigates gaseous release in process bay or process water tank room through HEPA filtration before discharge to outside of the facility and maintain DP within process bays</td>
<td>Airborne contamination within a process bay or process water tank room will be swept into the process general exhaust system, which includes HEPA filters and collects air from the process bays and process water tank. The system also provides confinement in conjunction with the facility structure by maintaining a negative building pressure</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Provides fail-closed exhaust dampers from process bays</td>
<td>Standby operation of local exhaust requires that other flow paths into the process bays are isolated to ensure DP in bays</td>
</tr>
<tr>
<td></td>
<td>Process bay recirculation HVAC system (HVACB system)</td>
<td>X</td>
<td>Provides fail-closed, outtake air inlet dampers so local exhaust on standby power can ensure process bay DP</td>
<td>(Design feature Section B5 6 11)</td>
</tr>
<tr>
<td></td>
<td>Reference air system components</td>
<td>X</td>
<td>Indicates building negative pressure demonstrating confinement is maintained by providing DP alarms and sufficient reference air connection</td>
<td>(Design feature Section B5 6 10)</td>
</tr>
<tr>
<td></td>
<td>Instrument air system</td>
<td>X</td>
<td>Provide safety-significant instrument air reservoir to operate process hood dampers during standby power operation of local exhaust system</td>
<td>Instrument air supply is GS Safety-significant reservoirs at each process hood damper ensure air supply to actuate pneumatic dampers for local exhaust operation on standby power</td>
</tr>
</tbody>
</table>

*SC - Safety Class, SS - Safety Significance
<table>
<thead>
<tr>
<th>Representative and bounding accident (Chapter B3 0)</th>
<th>Safety structures, systems and components</th>
<th>Safety function</th>
<th>Summary justification of safety function</th>
<th>Controls</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gaseous release (cont.)</td>
<td>Standby electrical power system</td>
<td>X</td>
<td>Provides standby power to maintain local exhaust flow to ensure sufficient DP in process bays during electrical outage. Provides restart capability to reestablish local exhaust flow to ensure sufficient DP after HVAC/PV system flow interruption.</td>
<td>Operation of local exhaust on standby power will ensure DP sufficient for confinement function during facility power outages. The standby power system provides direct-wired connections to restart the local exhaust fans and supporting equipment for the system to function on standby power.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Liquid release</td>
<td>CVDF structural features</td>
<td>X</td>
<td>Provide confinement for liquid sprays and aerosols and for retention of liquid pools in process water tank room.</td>
<td>The process water tank room floor slab walls and roof, in conjunction with the general exhaust system, provide confinement for liquid sprays and aerosol releases within the process water tank room. The tank room floor is designed to contain the entire liquid inventory of all tanks with elevated doorways. Lumped leaks in the bay were estimated less than 5 gal, which would not be safety significant.</td>
</tr>
<tr>
<td>(Section B3 4 2 2)</td>
<td>Process general supply/exhaust HVAC system (HVACD system)</td>
<td>X</td>
<td>Prevents radioactive materials released to the process water tank room from being released to the environment by collection via exhaust ductwork to exhaust HEPA filters.</td>
<td>Airborne contamination and aerosols within the process water tank room will be swept into the HVACD system, which includes HEPA filters. The general exhaust is the only facility ventilation in the process water tank room. The general exhaust provides confinement in conjunction with the facility structure by maintaining a negative building pressure. (Local exhaust does not serve the process water tank room, and leaks to process bays do not exceed guidelines.)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Reference air system components</td>
<td>X</td>
<td>Indicate building negative pressure during building construction and demonstrate confinement is maintained by providing DP alarms and sufficient reference air connection.</td>
<td>The reference air system provides SS DP indication and alarms to the control room for operator response. TSR to shut down process water pumps if DP is lost in process water tank room.</td>
</tr>
<tr>
<td></td>
<td>MCO external hydrogen explosion</td>
<td>CVDF structural features</td>
<td>X</td>
<td>Provide confinement for gaseous release inside process bays as a result of an external hydrogen explosion discharging into the process bay.</td>
</tr>
<tr>
<td>(Section B3 4 2 3)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>SCIC system</td>
<td>X</td>
<td>Detects high temperatures in process bay that could cause equipment malfunction during fire event.</td>
<td>The SCIC high bay temperature trip results in ISO &amp; PURGE alarm with required TSR response that protects assumptions of the FHA necessary to retain function of SC equipment.</td>
</tr>
</tbody>
</table>
Table B4-1  Safety-Class and Safety-Significant Structures, Systems and Components Summary List  (16 sheets)

<table>
<thead>
<tr>
<th>Representative and bounding accident (Chapter B3 0)</th>
<th>Safety structures, systems and components^1</th>
<th>Description</th>
<th>SC</th>
<th>SS</th>
<th>Safety function</th>
<th>Summary justification of safety function^2</th>
<th>Controls</th>
</tr>
</thead>
<tbody>
<tr>
<td>MCO external hydrogen explosion (cont.)</td>
<td>SCHe system</td>
<td>X</td>
<td></td>
<td></td>
<td>Upon actuation by loss of electrical power (or the SCIC system), provides pressure-regulated discharge flow path from the MCO to the local exhaust system</td>
<td>Although SCIC trip and SCHe actuation are not prevention actions for an external hydrogen explosion, if the SCHe is actuated, it has design features to preclude external hydrogen explosions. This includes the pressure regulators to the local exhaust.</td>
<td>LCO 3 2 1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Pressure-regulated discharge flow path from the MCO to the local exhaust system shall ensure at least a 1-minute delay from SCHe initiation until discharge flow initiation to local exhaust to allow standby power restart of local exhaust</td>
<td>The settings for flow rate and the SCHe supply and vent pressures ensure a design function time delay prior to venting to the local exhaust duct. That delay function will allow for restart of the standby power operation of the local exhaust prior to discharge of the ductwork such that sufficient dilution flow is available to preclude flammable mixtures in the duct.</td>
<td>LCO 3 2 1</td>
</tr>
<tr>
<td>Cask-MCO components</td>
<td></td>
<td>X</td>
<td></td>
<td></td>
<td>Provide pressure boundary for confinement of process gases</td>
<td>The MCO is the primary confinement boundary for the spent nuclear fuel. Process gases flow through the MCO</td>
<td>(Design features Sections B5 6 3 and B5 6 4)</td>
</tr>
<tr>
<td>VPS safety-class components</td>
<td></td>
<td>X</td>
<td></td>
<td></td>
<td>Provide fail-closed VPS isolation valves with IA line filters for MCO isolation pressure boundary</td>
<td>VPS isolation valves allow flow of process gases to the local exhaust ductwork. The isolation function is required when the SCHe is actuated.</td>
<td>LCO 3 2 1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Provide process piping and flex hoses for pressure boundary connection thru VPS to local exhaust duct</td>
<td>Flex hoses and piping connect to isolation valves and become an extended pressure boundary for flow of process gases to the local exhaust.</td>
<td>LCO 3 2 1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Provide connections for process lines to SCHe system piping</td>
<td>SCHe flow is through the VPS flex hoses</td>
<td>LCO 3 3 1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Provide process connectors for connection of process lines to MCO processing ports</td>
<td>Process connectors connect the MCO at the ports to the flex hoses. These must be leak tight and become part of the extended pressure boundary for the MCO</td>
<td>LCO 3 3 1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Provide 30 lb/in^2 vent path to local exhaust duct with sufficient line restriction to preclude flammable mixtures in operating local exhaust</td>
<td>The 30 lb/in^2 gauge vent path has design basis characteristics (line sizes due to diameter and length) that are credited to restrict flow rate to the local exhaust duct to preclude flammable mixtures from being generated in the local exhaust duct.</td>
<td>LCO 3 3 1</td>
</tr>
<tr>
<td>General-service helium system safety-class components</td>
<td></td>
<td>X</td>
<td></td>
<td></td>
<td>Provide isolation valves for pressure boundary connection for helium flow</td>
<td>An isolation valve in the open position allows flow of helium through MCO. The isolation function is required when the SCHe is actuated.</td>
<td>LCO 3 2 1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Provide fail-closed helium isolation valves with IA line filters for MCO isolation pressure boundary</td>
<td>The piping that connects general-service helium isolation valves to the VPS line become part of the pressure boundary for the MCO</td>
<td>LCO 3 2 1</td>
</tr>
</tbody>
</table>

^1 Safety structures, systems and components include those in the Safety Class or Safety Significant Category.

^2 Summary justification of safety function includes a description of how the safety function is achieved and any relevant design features.
<table>
<thead>
<tr>
<th>Representative and bounding accident (Chapter B3.0)</th>
<th>Safety structures, systems and components</th>
<th>Safety function</th>
<th>Summary justification of safety function</th>
<th>Controls</th>
</tr>
</thead>
<tbody>
<tr>
<td>MCO external hydrogen explosion (cont.)</td>
<td>PWC system safety-class components</td>
<td>X Provide isolation valves for pressure boundary connection and fail-closed PWC isolation valves with IA line filters for MCO isolation pressure boundary Provide process piping and flex hoses for pressure boundary connection and isolation of the PWC from MCO</td>
<td>Process water piping connects to the MCO long process tube port and contains process water isolation valves. The isolation function is required when the SCHe is actuated. Flex hoses and piping connect to the isolation valves and become an extended pressure boundary for the MCO such that the SCHe can function.</td>
<td>LCO 3 2 1</td>
</tr>
<tr>
<td>Process bay local exhaust HVAC and process vent system (HVACC/PV system)</td>
<td>X Maintain local exhaust flow sufficient to dilute potentially hydrogen-rich process gas discharges into exhaust duct to preclude flammable mixtures Reestablishes local exhaust flow sufficient to dilute potentially hydrogen-rich process gas discharges into exhaust duct to preclude flammable mixtures within 1 minute after HVACC/PV system loss of flow accident (with standby power interface) Provides hood isolation dampers Provides process hood low flow alarm to control room Mitigates gas cloud release in process bay as a result of a hydrogen explosion discharging into the process bay through HEPA filtration before discharge to outside of the facility With standby power interface, maintain a sufficient process bay DP when the HVACC system is not operable Provides cask venting connection with flow restricting orifice along with shut-off valve interlocked to local exhaust low flow switch</td>
<td>Exhaust flow of 1 000 ft³/min required for cask venting. Low flow alarm set above 1 070 ft³/min. Flow is also maintained during standby operation. SCHe design pressures provide for at least a 1-minute delay prior to discharging into the local exhaust, which must be running to dilute purge flow from MCO The restart function actuates dampers to reopen. The system is required to verify adequate dilution flow is available prior to cask venting. Some airborne contamination, if discharged within a bay as a result of an internal hydrogen explosion, will be swept into the local exhaust system, which includes HEPA filters and collects air from the process bays. The local exhaust in restart mode provides confinement in conjunction with the facility structure by maintaining a negative building pressure without the general exhaust or automatic temperature control operating. The flow restricting orifice is required to keep concentrations below the flammable limit even with 1 000 ft³/min flow in ductwork. Because flammable concentrations would be generated almost instantaneously during cask venting into a stagnant local exhaust duct, interlock of flow alarm to cask venting valve is required.</td>
<td>LCO 3 4 2</td>
<td></td>
</tr>
<tr>
<td>CVDF specialty tools</td>
<td>X Cask vent jumper and MCO vent jumper prevent hydrogen venting or leaking into process bay</td>
<td></td>
<td>The vent jumpers provide a reliable, leak-tight connection between the cask vent port or MCO and the process vent connection of the local exhaust system so hydrogen is not released uncontrolled to process bay which also provides worker safety protection.</td>
<td>(Design feature Section B5 6 1)</td>
</tr>
<tr>
<td>Representative and bounding accident (Chapter B3 0)</td>
<td>Safety structures, systems and components</td>
<td>Safety function(s)</td>
<td>Summary justification of safety function(s)</td>
<td>Controls</td>
</tr>
<tr>
<td>--------------------------------------------------</td>
<td>------------------------------------------</td>
<td>-------------------</td>
<td>---------------------------------------------</td>
<td>----------</td>
</tr>
<tr>
<td>MCO external hydrogen explosion (cont.)</td>
<td>Process general supply/exhaust HVAC system (HVACD system)</td>
<td>Prevents radioactive materials released to the process bay as a result of an external hydrogen explosion from being released to the environment by collection via exhaust ductwork to exhaust HEPA filters, maintains negative DP within process bays and process water tank room during normal HVAC operating conditions, provides fail-closed exhaust dampers from process bays</td>
<td>Airborne contamination within a bay will be swept into the process general exhaust system, which includes HEPA filters and collects air from the process bays.</td>
<td>LCO 3 4 1</td>
</tr>
<tr>
<td>Process bay recirculation HVAC system (HVACB system)</td>
<td>X</td>
<td>Provides fail-closed, outside air inlet dampers so local exhaust on standby power can ensure process bay DP</td>
<td>Standby operation of local exhaust requires other flow paths into the process bay are isolated to ensure DP in bays. This outside air inlet damper is large enough to ensure local exhaust system capability to maintain DP in standby mode</td>
<td>LCO 3 4 3 (Design feature Section B 5 6 11)</td>
</tr>
<tr>
<td>Reference air system components</td>
<td>X</td>
<td>Indicates building negative pressure required for confinement is maintained by providing DP alarm and sufficient reference air connection</td>
<td>The reference air system provides SS DP indication and alarms to the control room for operator response</td>
<td>LCO 3 4 3 (Design feature Section B 5 6 10)</td>
</tr>
<tr>
<td>Instrument air system</td>
<td>X</td>
<td>Provides safety-significant instrument air reservoir to operate process hood dampers during standby power operation of local exhaust system</td>
<td>Instrument air supply is GS. Safety-significant reservoirs at each process hood damper ensure air supply to actuate pneumatic dampers for local exhaust operation on standby power</td>
<td>LCO 3 4 2</td>
</tr>
<tr>
<td>Standby electrical power system</td>
<td>X</td>
<td>Provides standby power to CVDF to reestablish local exhaust flow to ensure sufficient DP in process bays during electrical outage, provides restart capability to reestablish local exhaust flow to ensure sufficient DP after HVAC/CS system flow interrupt</td>
<td>Operation of local exhaust on standby power will ensure DP sufficient for confinement function during facility power outages</td>
<td>LCO 3 5 1</td>
</tr>
<tr>
<td>MCO internal hydrogen explosion (Section B3 4 2.4)</td>
<td>CVDF structural features</td>
<td>X</td>
<td>Provides confinement for gas gases released inside process bays as a result of an internal hydrogen explosion discharging into the process bay</td>
<td>The CVDF process bays (2, 3, 4 and 5) floor slab walls, and roof, in conjunction with the HVAC exhaust systems, maintain DP and provide confinement for gas releases within the CVDF. An internal hydrogen explosion could blow out through a broken line into a bay. Both HVAC systems then mitigate the release</td>
</tr>
</tbody>
</table>
### Table B4-1  Safety-Class and Safety-Significant Structures, Systems and Components Summary List  (16 sheets)

<table>
<thead>
<tr>
<th>Representative and bounding accident (Chapter B3.6)</th>
<th>Safety structures, systems and components*</th>
<th>Safety function</th>
<th>Summary justification of safety function</th>
<th>Controls</th>
</tr>
</thead>
<tbody>
<tr>
<td>SCIC system</td>
<td>X</td>
<td>Monitors process and facility parameters, detects abnormal conditions, initiates MCO isolation and SCHe pressurization and purge</td>
<td>The system uses PLCs, wiring to process instrumentation, vacuum limit timer signals from seismic detectors and temperature sensors, system, controls, and output relays to isolate the MCO and activate the SCHe system or cut power to TW heater if abnormal process or environmental conditions are detected.</td>
<td>LCO 311</td>
</tr>
<tr>
<td>SCIC system</td>
<td>X</td>
<td>Detects seismic events that could impair SCIC function and activate MCO isolates and SCHe pressurization and purge</td>
<td>The SCIC seismic trip isolates MCO and activates SCHe such that seismic event cannot cause internal hydrogen explosion.</td>
<td>LCO 311 and LCO 317</td>
</tr>
<tr>
<td>SCHE system</td>
<td>X</td>
<td>Detects high temperatures in process that could cause instrument inaccuracies or malfunctions and initiates MCO isolation and SCHe pressurization and purge</td>
<td>The SCIC high temperature trip isolates MCO and activates SCHe such that excessive temperature cannot cause instrument inaccuracies or malfunctions that may initiate internal hydrogen explosion. The SCIC high temperature trip also protects assumptions of the FHA.</td>
<td>LCO 311 and LCO 316</td>
</tr>
<tr>
<td>SCHE system</td>
<td>X</td>
<td>Receipt of TW high water temperature signal to SCIC system initiates TW trip interlock, which shuts off power to TW heater</td>
<td>The SCIC TW trip precludes overheating the MCO due to a TW malfunction. Increased temperature would accelerate the hydrogen generation rate, which could exceed the design basis for hydrogen mitigation safety systems.</td>
<td>LCO 311</td>
</tr>
<tr>
<td>Cask-MCO safety-class components</td>
<td>X</td>
<td>Receipt of TW low water level in cask-MCO annulus signal to SCIC system activates low-level alarm, which initiates operator response</td>
<td>Loss of TW annulus water could also result in increased temperature within the MCO. SC alarms to the control room initiate operator response for recovery action.</td>
<td>LCO 311</td>
</tr>
<tr>
<td>Cask-MCO safety-class components</td>
<td>X</td>
<td>Upon activation by the SCIC system, prevents flammable concentrations of hydrogen and oxygen from forming within the MCO by pressurizing and purge functions</td>
<td>The SCHE system provides two redundant paths for passively purging and pressurizing the MCO and venting the process vent. The SCHe system is activated by the SCIC system. SCHe design flows and pressures prevent internal hydrogen explosion.</td>
<td>LCO 321</td>
</tr>
<tr>
<td>Cask-MCO safety-class components</td>
<td>X</td>
<td>Provide pressure boundary for confinement of process gases</td>
<td>The MCO is the primary confinement boundary for the spent nuclear fuel. The MCO closure contains plug valves and connections for the process piping to the SCHe system. The MCO outside surface interfaces with the annulus of its shielding/shipping cask to form an annulus through which annulus water is circulated by the TW</td>
<td>(Design features Sections B5 6.3 and B5 6.4)</td>
</tr>
<tr>
<td>Cask-MCO safety-class components</td>
<td>X</td>
<td>Provide connections for process lines to SCHe system piping</td>
<td>(Design features Sections B5 6.3 and B5 6.4)</td>
<td></td>
</tr>
<tr>
<td>Cask-MCO safety-class components</td>
<td>X</td>
<td>Provide hydraulic boundary connections and sealing surface for TW components to retain water in annulus for temperature control of MCO</td>
<td>(Design features Sections B5 6.3 and B5 6.4)</td>
<td></td>
</tr>
<tr>
<td>Representative and bounding accident (Chapter B3 6)</td>
<td>Safety structures, systems and components</td>
<td>Safety function</td>
<td>Summary justification of safety function</td>
<td>Controls</td>
</tr>
<tr>
<td>-----------------------------------------------</td>
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</tr>
<tr>
<td>MCO internal hydrogen explosion (cont.)</td>
<td>TW safety-class components</td>
<td>TW SC piping retains annulus water via piping integrity and annulus valves</td>
<td>A portion of the system contains safety-class piping and annulus valves to retain a minimum water level above the elevation of the fuel within the MCO</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Detect low water level in cask-MCO annulus and provide signal to SCIC system to actuate low-level alarm</td>
<td>Redundant SC level indicators are used in conjunction with the SCIC to demonstrate water level is maintained. Passive presence of water in annulus is the safety function for heat conduction. The TW level check petcock can be used to verify alarm.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Detect high water temperature and provide signal to SCIC system to TW trip interlock</td>
<td>The SCIC TW trip precedes overheating the MCO due to a TW heater control malfunction. Increased temperature would accelerate the hydrogen generation rate that could exceed the design basis for hydrogen mitigation safety systems</td>
<td></td>
</tr>
<tr>
<td>VPS safety-class components</td>
<td>X</td>
<td>Provide fail-closed VPS isolation valves with 1A lane filters for MCO isolation pressure boundary</td>
<td>One connection of the VPS piping connects to the MCO process port and contains VPS isolation valves that are closed by the SCIC system. Air supply at GOVs includes SC filter to preclude foreign material in actuator</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Provide process piping and flex hoses to isolation valves for MCO isolation pressure boundary</td>
<td>Another connection of the VPS piping connecting to the MCO long process tube port contains deoxygenized water isolation valves that are closed by the SCIC system. Air supply at GOVs includes SC filter to preclude foreign material in actuator</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Provide process connectors for connection of process lines to MCO processing ports</td>
<td>Flex hoses and piping connect to isolation valves and become an extended pressure boundary for the MCO such that the SCIC can function.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Provide MCO pressure information to SCIC system to initiate MCO isolation and SCIC actuation during process upset conditions</td>
<td>Process connectors connect the MCO to the ports to the flex hoses. That must be leaktight and become part of the extended pressure boundary for the MCO</td>
<td></td>
</tr>
<tr>
<td>General-service helium system safety-class components</td>
<td>X</td>
<td>Provide helium flow information to SCIC system to initiate MCO isolation and SCIC actuation during process upset conditions</td>
<td>Helium purge flow as a key process parameter. SC flowmeters interface with the SCIC system and monitor the helium flow in the process.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Provide fail-closed general-service helium isolation valves with 1A lane filters for MCO isolation pressure boundary</td>
<td>The general-service helium connects to the VPS piping connection to the MCO process port. General-service helium isolation valves become part of the extended MCO pressure boundary. Air supply lines to the GOV actuators include SC filters to preclude foreign material in actuator</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Provide process piping to isolation valves for MCO isolation pressure boundary</td>
<td>The piping that connects general-service helium isolation valves to the VPS line becomes part of the extended pressure boundary for the MCO</td>
<td></td>
</tr>
</tbody>
</table>

*Design feature Section B5 6.2*
<table>
<thead>
<tr>
<th>Representative and bounding accident (Chapter B3 0)</th>
<th>Safety structures, systems and components</th>
<th>Safety function</th>
<th>Summary justification of safety function</th>
<th>Controls</th>
</tr>
</thead>
<tbody>
<tr>
<td>MCO internal hydrogen explosion (cont.)</td>
<td>PWC system safety-class components</td>
<td>X</td>
<td>Provide fail-closed process water isolation valves with IA line filters for MCO isolation pressure boundary</td>
<td>LCO 3 2 1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Provide process piping and flex hose to isolation valves for MCO isolation pressure boundary</td>
<td>LCO 3 2 1</td>
</tr>
<tr>
<td>Process bay local exhaust HVAC and process vent system (HVACC/PV system)</td>
<td>X</td>
<td>Mitigates gaseous release in process bay as a result of an internal hydrogen explosion discharging into the process bay through HEPA filtration before discharge to outside of the facility</td>
<td>Some airborne contamination, if discharged within a bay through a broken line as a result of an internal hydrogen explosion, will be swept into the local exhaust system, which includes HEPA filters and collects air from the process bays</td>
<td>LCO 3 4 2</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Recaptures local exhaust flow after HVAC/PV system loss of flow accident (with standby power interface)</td>
<td>LCO 3 4 2</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Provides hood isolation dampers</td>
<td>LCO 3 4 2</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>With standby power interface, maintains a sufficient process bay DP when the process HVAC system is not operable</td>
<td>LCO 3 4 2</td>
</tr>
<tr>
<td>Process general supply/exhaust HVAC system (HVACD system)</td>
<td>X</td>
<td>Prevents radioactive materials released to the process bay as a result of an internal hydrogen explosion from being released to the environment by collection via exhaust ductwork to exhaust HEPA filters</td>
<td>Airborne contamination within a bay will be swept into the process general exhaust system, which includes HEPA filters and collects air from the process bays</td>
<td>LCO 3 4 1</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Maintains negative DP within process bays during normal HVAC operating conditions</td>
<td>LCO 3 4 3</td>
</tr>
<tr>
<td></td>
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<td></td>
<td>Provides fail-closed exhaust dampers from process bays</td>
<td>Design feature Section B5 6 11</td>
</tr>
<tr>
<td>Process bay recirculation HVAC system (HVACB system)</td>
<td>X</td>
<td>Provide fail-closed outside air inlet dampers so local exhaust on standby power can ensure process bay DP</td>
<td>Standby operation of local exhaust requires other flow paths into the process bay are isolated to ensure DP in bays</td>
<td>Design feature Section B5 6 10</td>
</tr>
<tr>
<td>Reference air system components</td>
<td>X</td>
<td>Indicate building negative pressure required for confinement as maintained by providing DP alarms and sufficient reference air connection</td>
<td>The reference air system provides SS DP indication and alarms to the control room for operator response</td>
<td>LCO 3 4 3</td>
</tr>
<tr>
<td>Instrument air system</td>
<td>X</td>
<td>Provide safety-significant instrument air reservoir to operate process hood dampers during standby power operation of local exhaust system</td>
<td>Instrument air supply is GS. Safety-significant reserves at each process hood damper ensure air supply to actuate pneumatic dampers for local exhaust operation on standby power</td>
<td>LCO 3 4 2</td>
</tr>
</tbody>
</table>
### Table B4-1 Safety-Class and Safety-Significant Structures, Systems and Components Summary List (16 sheets)

<table>
<thead>
<tr>
<th>Representative and bounding accident (Chapter B3 0)</th>
<th>Safety structures, systems and components</th>
<th>Safety function</th>
<th>Summary justification of safety function</th>
<th>Controls</th>
</tr>
</thead>
<tbody>
<tr>
<td>MCO thermal runaway reaction (Section B3 4 2 5)</td>
<td>SCIC system</td>
<td>X</td>
<td>Provides standby power to CVDF to restore local exhaust flow to ensure sufficient DP in process bays during electrical outage. Provides restart capability to reestablish local exhaust flow to ensure sufficient DP after HVAC/FA system flow interrupt.</td>
<td>LCO 3 1 1</td>
</tr>
<tr>
<td></td>
<td>SCIC system</td>
<td>X</td>
<td>The system uses PLCs, wiring to process instrumentation, signals from system detectors and temperature monitors, system controls, and output relays to isolate the MCO and actuate the SCIC system or cut power to TW heater if abnormal process or environmental conditions are detected.</td>
<td>LCO 3 1 1 and LCO 3 1 7</td>
</tr>
<tr>
<td></td>
<td>SCIC system</td>
<td>X</td>
<td>The SCIC system will isolate the MCO and actuate the SCIC system that it is a new event cannot initiate thermal runaway due to equipment malfunction.</td>
<td>LCO 3 1 1 and LCO 3 1 6</td>
</tr>
<tr>
<td></td>
<td>SCIC system</td>
<td>X</td>
<td>The SCIC high bay temperature trip isolates the MCO and actuates the SCIC system that if excessive temperature cannot cause instrument malfunctions or malfunctions that may initiate thermal runaway.</td>
<td>LCO 3 1 1</td>
</tr>
<tr>
<td></td>
<td>SCIC system</td>
<td>X</td>
<td>The SCIC system will actuate system to prevent the MCO due to a TW malfunction. Increased temperature would accelerate the corrosion reaction and generate additional heat initiating a thermal runaway.</td>
<td>LCO 3 1 1</td>
</tr>
<tr>
<td></td>
<td>SCIC system</td>
<td>X</td>
<td>Loss of TW annulus water could also result in increased temperature within the MCO increasing corrosion rates SC alarms to the control room, actuate operator response for recovery action.</td>
<td>LCO 3 1 1</td>
</tr>
<tr>
<td></td>
<td>SCIC system</td>
<td>X</td>
<td>Inadvertent water ingress during the drying and proof tests could result in process upset that could initiate thermal runaway if unmitigated at CVDF.</td>
<td>LCO 3 1 1</td>
</tr>
<tr>
<td></td>
<td>SCIC system</td>
<td>X</td>
<td>The SCIC system provides two redundant paths for passively purging and pressurizing the MCO and venting to the local exhaust. The SCIC system is actuated by the SCIC system. SCIC design flows and pressures provide a pressure vent path to the local exhaust system.</td>
<td>LCO 3 2 1</td>
</tr>
</tbody>
</table>

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<table>
<thead>
<tr>
<th>Representative and bounding accident (Chapter B3.0)</th>
<th>Safety structures, systems and components</th>
<th>Safety function</th>
<th>Summary justification of safety function</th>
<th>Controls</th>
</tr>
</thead>
<tbody>
<tr>
<td>MCO thermal runaway reactions (cont.)</td>
<td>Cask-MCO components</td>
<td>Provide pressure boundary for confinement of process gases</td>
<td>The MCO is the primary confinement boundary for the spent nuclear fuel</td>
<td>(Design feature Sections B5 6.3 and B5 6.4)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Provide connections for process lines to SChE system piping</td>
<td>The MCO closure contains plug valves and connections for the process piping to the SChE system.</td>
<td>(Design feature Sections B5 6.3 and B5 6.4)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Provide hydraulic boundary SC connections, and sealing surface for TW components to retain water in the cask-MCO annulus</td>
<td>The MCO outside surface interfaces with the mode of its shielding/shipping cask to form an annulus through which annulus water is supplied by the TW. Lower cask port fitting must be SC for TW annulus water connection.</td>
<td>(Design feature Sections B5 6.3 and B5 6.4)</td>
</tr>
<tr>
<td>TW safety-class components</td>
<td>X</td>
<td>TW SC piping retain annulus water via piping integrity and antiphase valves</td>
<td>A portion of the system contains SC piping and antiphase valves to retain a minimum water level above the elevation of the fuel within the MCO</td>
<td>(Design feature Section B5 6.2)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Detect low water level in cask-MCO annulus and provide signal to SCIC system to activate low-level alarm</td>
<td>Redundant SC liquid level indicators are used in conjunction with the SCIC to demonstrate water level is maintained. Passive presence of water in annulus is the safety function for heat conduction. The TW level check potentiometers can be used to verify alarm.</td>
<td>LCO 3.1.5</td>
</tr>
<tr>
<td></td>
<td></td>
<td>System design provides manual refill capability</td>
<td>Manual refill capability could be required if low annulus water level is detected. Loss of annulus water would require multiple failures</td>
<td>(Design feature Section B5 6.2)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Detect high-water supply temperature and provide signal to SCIC system to TW trip interlock</td>
<td>The SCIC TW trip prescribes overheating the MCO due to a TW heater control malfunction. Increased temperature would accelerate corrosion rate, which could induce thermal runaway</td>
<td>LCO 3.1.4</td>
</tr>
<tr>
<td>VPS safety-class components</td>
<td>X</td>
<td>Provide fail-closed VPS isolation valves with IA line filters for MCO isolation pressure boundary</td>
<td>One connection of the VPS piping connects to the MCO process port and contains VPS isolation valves that are closed by the SCIC system. Air supply to GVs includes SC filter to preclude foreign material in actuator</td>
<td>LCO 3.2.1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Provide fail-closed deionized water isolation valves with IA line filters for MCO isolation pressure boundary</td>
<td>Another connection of the VPS piping connecting to the MCO long process tube port contains deionized water isolation valves that are closed by the SCIC system. Air supply at GVs includes SC filter to preclude foreign material in actuator</td>
<td>LCO 3.2.1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Provide process piping and flex hoses to isolation valves for MCO isolation pressure boundary</td>
<td>Flex hoses and piping connect to isolation valves and become an extended pressure boundary for the MCO such that the SChE can function.</td>
<td>LCO 3.2.1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Provide process connectors for connection of process lines to MCO processing ports</td>
<td>Process connectors connect the MCO at the ports to the flex hoses. Tensile testing and become part of the extended pressure boundary for the MCO</td>
<td>LCO 3.3.1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Provide MCO pressure information to SCIC system to initiate MCO isolation and SChE actuation during process upset conditions</td>
<td>Pressure indicators are monitored by the SCIC system to monitor the MCO internal pressure. PLC logic programs monitor acceptable pressure parameters based on SCIC mode that, if exceeded, could indicate process upset conditions that are terminated by SCIC system</td>
<td>LCO 3.1.2</td>
</tr>
<tr>
<td>Representative and bounding accident (Chapter B3 0)</td>
<td>Safety structures, systems and components*</td>
<td>Safety function</td>
<td>Summary justification of safety function b</td>
<td>Controls</td>
</tr>
<tr>
<td>--------------------------------------------------</td>
<td>-------------------------------------------</td>
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<td>------------------------------------------</td>
<td>----------</td>
</tr>
<tr>
<td>MCO thermal runaway reactions (cont.)</td>
<td>General-service helium system safety-class components</td>
<td>X</td>
<td>Provide fail-closed general-service helium isolation valves with 1A line filters for MCO isolation pressure boundary</td>
<td>The general-service helium connects to the VFS piping connection to the MCO process port. General-service helium isolation valves become part of the extended MCO pressure boundary. Air supply at GOVs includes SC filter to preclude foreign material in actuator.</td>
</tr>
<tr>
<td>PWC system safety-class components</td>
<td>Provide fail-closed process water isolation valves with 1A line filters for MCO isolation pressure boundary</td>
<td>X</td>
<td>Provide process piping to isolation valves for MCO isolation pressure boundary</td>
<td>The piping that connects general-service helium isolation valves to the VFS line become part of the extended pressure boundary for the MCO.</td>
</tr>
<tr>
<td>Process bay local exhaust HVAC and process vent system (HVACC/PV system)</td>
<td>Provide process piping and field hoses to isolation valves for MCO isolation pressure boundary</td>
<td>X</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Process bay local exhaust HVAC and process vent system (HVACC/PV system)</td>
<td>Process water piping connects to the MCO long process tubes port and contains process water isolation valves that are closed by the SCIC system. Air supply at GOVs includes SC filter to preclude foreign material in actuator.</td>
<td></td>
<td></td>
<td>LCO 3 21</td>
</tr>
<tr>
<td>Process bay local exhaust HVAC and process vent system (HVACC/PV system)</td>
<td>Field hoses and piping connect to the isolation valves and becomes an extended pressure boundary for the MCO such that the SCHe can function.</td>
<td></td>
<td></td>
<td>LCO 3 21</td>
</tr>
<tr>
<td>Instrument air system</td>
<td>Provide safety-significant instrument air reservoir to operate process hood dampers during standby power operation of local exhaust system</td>
<td>X</td>
<td>Instrument air supply is GS. Safety-significant reservoirs at each process hood damper ensure air supply to actuate pneumatic dampers for local exhaust operation on standby power</td>
<td>LCO 3 4 2</td>
</tr>
<tr>
<td>Standby electrical power system</td>
<td>Provides standby power to CVDF to reestablish local exhaust flow to ensure sufficient DP as process bays during electrical outage</td>
<td>X</td>
<td>Operation of local exhaust on standby power will ensure DP sufficient for confinement function during facility power outages.</td>
<td>LCO 3 5 1</td>
</tr>
<tr>
<td>Standby electrical power system</td>
<td>Provides restart cavities to reestablish local exhaust flow to ensure sufficient DP after HVACC/PV system flow interrupt</td>
<td></td>
<td>The standby power system provides direct wired connections to restart the local exhaust fans and supporting equipment for system to function on standby power</td>
<td>LCO 3 5 1</td>
</tr>
<tr>
<td>Safety Class and Safety-Significant Structures, Systems and Components Summary List (16 sheets)</td>
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</tr>
<tr>
<td><strong>Representative and bounding accident</strong> (Chapter B3 0)</td>
<td><strong>Safety structures, systems and components</strong></td>
<td><strong>Safety function</strong></td>
<td><strong>Summary justification of safety function</strong></td>
<td><strong>Controls</strong></td>
</tr>
<tr>
<td>MCO overpressurization</td>
<td>CVDF structural features</td>
<td><strong>X</strong></td>
<td>Provide confinement for potential gaseous release inside process bay as a result of the 150 lb/in² rupture disk discharging into the process bay</td>
<td>The CVDF process bays (2, 3, 4 and 5) floor slab walls, and roof, in conjunction with the HVAC exhaust systems, maintain DP and provide confinement for gaseous releases within the CVDF. A blowdown through the 150 lb/in² rupture disk would release to the process bay. Both HVAC systems then mitigate the release.</td>
</tr>
<tr>
<td>SCIC system</td>
<td></td>
<td><strong>X</strong></td>
<td>Monitors process and facility parameters and detects abnormal conditions that initiate MCO isolation and SCHe pressurization and purge</td>
<td>The system uses PLCs, warning to process instrumentation, signals from seismic detectors and temperature monitors, system controls, and output relays to isolate the MCO and activate the SCHe system or cut power to TW heater if abnormal process or environmental conditions are detected.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Detects seismic events that could impair SCIC function and initiate MCO isolation and SCHe pressurization and purge</td>
<td>The SCIC seismicrop isolates MCO and activates SCHe such that seismic event cannot initiate MCO overpressurization due to equipment malfunction</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Detects high temperatures in process bay that could cause instrument maccruptcy or malfunction and initiate MCO isolation and SCHe pressurization and purge</td>
<td>The SCIC high bay temperature trip isolates MCO and activates SCHe such that excessive temperature cannot cause instrument maccruptcy or malfunctions that may initiate overpressurization. The SCIC high bay temperature trip also protects assumptions of the FHA.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Receipt of TW high water temperature signal to SCIC system initiates TW trip interlock that shuts off power to TW heater</td>
<td>The SCIC TW trip precludes overheating the MCO due to a TW malfunction. Increased temperature would accelerate the corrosion reaction and generate additional heat mutating an overpressurization.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Receipt of TW low water level in B-1 cool-MCO annulus signal to SCIC system actuates low-level alarm that initiates operator response</td>
<td>Loss of TW annulus water could also result in decreased temperature within the MCO increasing corrosion rates. SC alarm to the control room initiate operator response for recovery action.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Interlocks PWC and deoxygenated water valves to preclude inadvertent water ingress to the MCO during drying or proof modes</td>
<td>Inadvertent water ingress during the drying or proof modes could result in process upset that could result in overpressurization consequences at the CVDF or CSB.</td>
</tr>
<tr>
<td>SCHe system</td>
<td></td>
<td><strong>X</strong></td>
<td>Upon actuation by the SCIC system, provides pressure regulated discharge flow path from the MCO to the local exhaust system</td>
<td>The SCHe system provides two redundant paths for venting the MCO to the local exhaust. The SCHe system is actuated by the SCIC system. SCHe design flows and pressures provide a pressure vent path to the local exhaust system.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Pressure control valve and rupture disk in each tank protect MCO from helium overpressurization from the He bottles.</td>
<td>The SCHe system provides two redundant paths for passively purging and pressurizing the MCO. Each purge system contains helium supplies at high pressure. Pressure control valves are provided along with SC rupture disks on each line from each bottle to ensure bottle pressure cannot be applied to the MCO. Bottle pressure of ~2,500 lb/in² gauge would exceed the 150 lb/in² gauge design pressure of the MCO and breach the confinement boundary. This is prescriptive a SC function to protect the MCO pressure boundary.</td>
</tr>
<tr>
<td>Annex B - Cold Vacuum Drying Facility</td>
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</table>

Table B4-1  Safety-Class and Safety-Significant Structures, Systems and Components Summary List  (16 sheets)

<table>
<thead>
<tr>
<th>Representative and bounding accident (Chapter B3 I)</th>
<th>Safety structures, systems and components</th>
<th>Safety function</th>
<th>Summary justification of safety function</th>
<th>Controls</th>
</tr>
</thead>
<tbody>
<tr>
<td>MCO overpressurization (cont.)</td>
<td>Cask-MCO components</td>
<td>Provide pressure boundary for confinement of process gases</td>
<td>The MCO is the primary confinement boundary for the spent nuclear fuel.</td>
<td>(Design feature: Sections B5 6 3 and B5 6 4)</td>
</tr>
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<td></td>
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<td>Provide 150 lb/in² gauge rupture disk</td>
<td>A 150 lb/in² gauge rupture disk is redundant to 30 lb/in² gauge path.</td>
<td>(Design feature: Sections B5 6 3 and B5 6 4)</td>
</tr>
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<td></td>
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<td>Provide connections for process lines to SCHx system piping</td>
<td>The MCO closure contains plug valves and connections for the process piping to the SCHx system.</td>
<td>(Design feature: Sections B5 6 3 and B5 6 4)</td>
</tr>
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<td></td>
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<td>Provide hydraulic boundary SC connections, and sealing surface for TW components to retain water in annulus for temperature control of MCO</td>
<td>The MCO outside surface interfaces with the core of the shielding/shipping cask to form an annulus through which annulus water is circulated by the TW. Drain fitting must be SC as TW annulus water connection.</td>
<td>(Design feature: Sections B5 6 3 and B5 6 4)</td>
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<td></td>
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<td>Provide pressure boundary with a design pressure of at least 150 lb/in²</td>
<td>The MCO provides the primary confinement of the spent nuclear fuel. This boundary has a design pressure of 150 lb/in² gauge.</td>
<td>(Design feature: Sections B5 6 3 and B5 6 4)</td>
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<td></td>
<td>TW safety-class components</td>
<td>TW SC piping retains annulus water via piping integrity and annulus valves</td>
<td>A portion of the system contains SC piping, annulus valves to retain a minimum water level above the elevation of the fuel within the MCO.</td>
<td>(Design feature: Sections B5 6 3 and B5 6 4)</td>
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<td>Detect low water level in cask-MCO annulus and provide signal to SCIC system to activate low-level alarm</td>
<td>Redundant SC fuel level indicators switch used in conjunction with the SCIC demonstrate water level is maintained. Passive presence of water in the annulus is the safety function for heat conduction. The TW level check petcock can be used to verify alarm.</td>
<td>LCO 3 1 4</td>
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<td></td>
<td>Detect high water supply temperature and provide signal to SCIC system TW trip interlock</td>
<td>The SCIC TW trip precludes overheating the MCO due to a TW heater control malfunction. Increased temperature would accelerate corrosion rate which could cause overpressurization.</td>
<td>LCO 3 1 5</td>
</tr>
<tr>
<td>Representative and bounding accident (Chapter B3.0)</td>
<td>Safety structures, systems and components</td>
<td>Safety function</td>
<td>Summary justification of safety function (^1)</td>
<td>Controls</td>
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<tr>
<td>MCO overpressurization (cont.)</td>
<td>VPS safety-class components</td>
<td>Provide fail-closed VPS isolation valves with 1A line filters for MCO isolation pressure boundary</td>
<td>One connection of the VPS piping connects to the MCO process port and contains VPS isolation valves that are closed by the SCIC system. Air supply at GOVs includes an SC filter to preclude foreign material in the actuator.</td>
<td>LCO 3 2 1</td>
</tr>
<tr>
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<td></td>
<td>Provide fail-closed deaerated water isolation valves with 1A line filters for MCO isolation pressure boundary</td>
<td>Another connection of the VPS piping connecting to the MCO long process tube port contains deaerated water isolation valves that are closed by the SCIC system. Air supply at GOVs includes SC filter to preclude foreign material in actuator.</td>
<td>LCO 3 2 1</td>
</tr>
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<td>Provide process piping and flex hoses to isolation valves for MCO isolation pressure boundary</td>
<td>Flex hoses and piping connect to isolation valves and become an extended pressure boundary for the MCO such that SCICs can function.</td>
<td>LCO 3 2 1</td>
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<td>Provide process connections for connection of process lines to MCO processing ports</td>
<td>Process connections connect the MCO at the ports to the flex hoses. This must be leaktight and become part of the extended pressure boundary for the MCO.</td>
<td>LCO 3 3 1</td>
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<td></td>
<td>Provide MCO pressure information to SCIC system to initiate MCO isolation and SCICs actuation during upset conditions</td>
<td>The SCIC system has a high pressure trip based on the MCO external pressure. PLC logic programs monitor acceptable pressure parameters based on SCIC mode. If exceeded, indicate process upset conditions that are terminated by SCIC trip.</td>
<td>LCO 3 1 2</td>
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<td>Provide 30 lb/ft(^2) vent path to local exhaust duct</td>
<td>The 30 lb/ft(^2) gauge vent path precludes high pressures and protects against venting through the 150 lb/ft(^2) gauge rupture disk. Discharges to local exhaust system are mitigated through HEPA filters.</td>
<td>LCO 3 3 1</td>
</tr>
<tr>
<td>General-service helium system safety-class components</td>
<td>VPS safety-class components</td>
<td>Provide fail-closed general-service helium isolation valves with 1A line filters for MCO isolation pressure boundary</td>
<td>The general-service helium connects to the VPS piping connection to the MCO process port. General-service helium isolation valves become part of the extended MCO pressure boundary. Air supply at GOVs includes an SC filter to preclude foreign material in actuator.</td>
<td>LCO 3 2 1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Provide process piping to isolation valves for MCO isolation pressure boundary</td>
<td>The piping that connects general-service helium isolation valves to the VPS line become part of the extended pressure boundary for the MCO.</td>
<td>LCO 3 2 1</td>
</tr>
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<td>Redundant pressure relief valves on supply line protect MCO from helium overpressurization from the helium tube trailers</td>
<td>The general-service helium system supplies helium from tube transporters at 2,600 to 3,200 lb/ft(^2) gauge. The main supply header redundant pressure relief valves (25 lb/ft(^2) gauge) are relied upon to prevent the MCO and related piping from experiencing full supply pressure in the event of regulator failure.</td>
<td>LCS/LCO 3 3 2</td>
</tr>
<tr>
<td>Table B4-1  Safety-Class and Safety-Significant Structures, Systems and Components Summary List (16 sheets)</td>
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<tr>
<td>Representative and bounding accident (Chapter B3 0)</td>
<td>Safety structures, systems and components*</td>
<td>Safety function</td>
<td>Summary justification of safety function b</td>
<td>Controls</td>
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<tr>
<td>MCO overpressurization (cont.)</td>
<td>PWC system safety-class components</td>
<td>X</td>
<td>Provide fail-closed process water isolation valves with 1A line filters for MCO isolation pressure boundary</td>
<td>Process water piping connects to the MCO long process tube port and contains process water isolation valves that are closed by the SCIC system. Air supply at GOV's includes SC filter to preclude foreign material in actuator LCO 3 2 1</td>
</tr>
<tr>
<td>Process bay local exhaust HVAC and process vent system (HVACC/PV system)</td>
<td>X</td>
<td>Mitigates gaseous release from SCHe vent and 30 lb/in³ gauge vent path and in process bay as a result of a 150 lb/in³ rupture disk discharging into the process bay through HEPA filtration before discharge via stack to mitigate onsite consequences Provides hood isolation dampers Provides process hood low flow alarm to control room Restablishes local exhaust flow after HVACC/PV system loss of flow accident (with standby power interface) With standby power interface, maintains a sufficient process bay DP when the HVACD system is not operable</td>
<td>Gaseous release from SCHe vent and the 30 lb/in³ gauge vent path along with release to the process bay from the 150 lb/in³ gauge rupture disk will be swept into the local exhaust system which includes HEPA filters and collects air from the process bays The restart function actuates dampers to re-open. The system is required to verify adequate dilution flow is available during 30 lb/in³ gauge venting. SCHe design pressures provide for at least a 1-minute delay prior to discharging into the local exhaust, which must be running to dilute purge flow from MCO The local exhaust in restart mode provides confinement in conjunction with the facility structure by maintaining a negative building pressure without the general exhaust or automatic temperature control operating.</td>
<td>LCO 3 4 2</td>
</tr>
<tr>
<td>Process general supply/exhaust HVAC system (HVACD system)</td>
<td>X</td>
<td>Prevents radioactive materials released to the process bay from MCO 150 lb/in³ rupture disk being released to the environment by collection via exhaust ductwork to exhaust HEPA filters to mitigate onsite consequences Maintains confinement negative DP within process bays during normal HVAC operating conditions Provides fail-closed exhaust dampers from process bays and process water tank room.</td>
<td>Airborne contamination within a bay will be swept into the process general exhaust system, which includes HEPA filters and collects air from the process bays The system provides confinement in conjunction with the facility structure by maintaining a negative building pressure Standby operation of local exhaust requires other flow paths into the process bay are isolated to ensure DP in bays.</td>
<td>LCO 3 4 1</td>
</tr>
<tr>
<td>Process bay recirculation HVAC system (HVACBI system)</td>
<td>X</td>
<td>Provides fail-closed outside air inlet dampers so local exhaust on standby power can ensure process bay DP</td>
<td>Standby operation of local exhaust requires other flow paths into the process bay are isolated to ensure DP in bays. This outside air inlet damper is large enough to unpair local exhaust system capability to maintain DP in standby mode</td>
<td>LCO 3 4 3</td>
</tr>
<tr>
<td>Reference air system components</td>
<td>X</td>
<td>Indicate building negative pressure required for confinement is maintained by providing DP alarms and sufficient reference air connection</td>
<td>The reference air system provides SS DP indication and alarms to the control room for operator response</td>
<td>LCO 3 4 3</td>
</tr>
<tr>
<td>Representative and bounding accident (Chapter B3.0)</td>
<td>Safety structures, systems and components*</td>
<td>Safety function</td>
<td>Summary justification of safety function b</td>
<td>Controls</td>
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<tr>
<td>MCO overpressurization (cont.)</td>
<td>Instrument air system</td>
<td>SC</td>
<td>Instrument air supply is GS. Safety-significant reservoirs at each process hood damper ensure air supply to activate pneumatic dampers for local exhaust operation on standby power</td>
<td>LCO 342</td>
</tr>
<tr>
<td></td>
<td>Standby electrical power system</td>
<td>SC</td>
<td>Operation of local exhaust on standby power will ensure DP sufficient for confinement function during facility power outages</td>
<td>LCO 351</td>
</tr>
<tr>
<td>Nuclear criticality</td>
<td>Cask-MCO safety-class components</td>
<td>SC</td>
<td>The standby power system provides direct-wired connections to restart the local exhaust fans and supporting equipment for system to function on standby power</td>
<td>LCO 351</td>
</tr>
</tbody>
</table>

*Safety function identifies safety-significant or safety-class function required to prevent or mitigate the specific accident. Safety-class components often perform safety-significant functions.

Summary system description provides overall system discussion. Table B4-2 text provides specific discussions of safety components.

*General-service structure designed and built to performance category 3 criteria provides safety-significant confinement function in conjunction with HEPA filtered exhaust systems.

CVDF = Cold Vacuum Drying Facility
DP = differential pressure
FHIA = fire hazards analysis
GUV = gas operating valve
GS = general service
HEPA = high-efficiency particulate air (filter)
HVAC = heating, ventilation, and air conditioning.
HVACB = process bay recirculation heating, ventilation, and air conditioning.
HVAC/PV = process bay heating, ventilation, and air conditioning and process vent.
HVACD = process general supply heating, ventilation, and air conditioning.
IA = instrument air
LCO = Limiting Condition of Operation.
MCO = mult-carrier overpack.
PLC = programmable logic controller
FWC = process water conditioning.
SC = safety class
SCHE = safety-class helium.
SCIC = safety-class instrumentation and control
SS = safety significant.
TSR = technical safety requirement.
TW = tempered water (auxiliary) system.
VPS = vacuum purge system.

Annex B — Cold Vacuum Drying Facility

November 1999
Table B4-2  Structure, System, and Component Safety Function Classification Summary (4 sheets)

<table>
<thead>
<tr>
<th>Structures systems and components</th>
<th>Gaseous release</th>
<th>Liquid release</th>
<th>External hydrogen explosion</th>
<th>Internal hydrogen explosion</th>
<th>Thermal runaway reaction</th>
<th>MCO overpressurization</th>
<th>Criticality prevention</th>
<th>NRC important to safety classification</th>
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<tbody>
<tr>
<td>Cold Vacuum Drying Facility structure</td>
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<td>Process bays (confinement)</td>
<td>GS(^1)</td>
<td>GS(^1)</td>
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<td>GS(^1)</td>
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<td>Process water tank room (confinement)</td>
<td>GS(^3)</td>
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<td>Liquid release</td>
<td>External hydrogen explosion</td>
<td>Internal hydrogen explosion</td>
<td>Thermal runaway reaction</td>
<td>MCO overpressurization</td>
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<td>Process water transfer line in process bays</td>
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<tr>
<td>HEPA filter</td>
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<tr>
<td>Ductwork</td>
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<td>Hood isolation damper (fail closed)</td>
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<tr>
<td>Cask venting orifice</td>
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<tr>
<td>Cask vent jumper tool</td>
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<td>Cask venting valve</td>
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<td>Cask venting flow interlock</td>
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<td>MCO vent jumper tool</td>
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<td>Process general supply/exhaust HVAC system</td>
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<td>Exhaust ductwork</td>
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<td>SS</td>
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<tr>
<td>Isolation dampers (process bays)</td>
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<td>SS</td>
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<td>Process bay recirculation HVAC system</td>
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<td>Isolation dampers (outside air inlet)</td>
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<td>Reference air system</td>
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<td>Reference air header</td>
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<tr>
<td>Differential pressure alarms</td>
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<td>Instrument air system</td>
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<td>Instrument air reservoirs</td>
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<td>Air reservoir pressure gauges</td>
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<tr>
<td>Structures systems and components</td>
<td>Gaseous release</td>
<td>Liquid release</td>
<td>External hydrogen explosion</td>
<td>Internal hydrogen explosion</td>
<td>Thermal runaway reaction</td>
<td>MCO overpressurization</td>
<td>Criticability prevention</td>
<td>NRC important to safety classification</td>
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<td>Standby electrical power</td>
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<tr>
<td>Diesel generator</td>
<td>SS</td>
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<tr>
<td>Local exhaust restart circuit</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
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</tbody>
</table>

Note: Safety classification of structures systems or components is listed as the highest designation for the specific design basis accident.

1. General service structure designed and built to performance category 3 criteria provides safety significant confinement function in conjunction with HEPA filtered exhaust system.
2. Designed to important to safety Category A for natural phenomena important to safety Category B for confinement functions and important to safety Category C for weather protection and operating environment control.
3. General service structure designed and built to performance category 2 criteria provides safety significant confinement function in conjunction with HEPA filtered exhaust system.
4. Designated as important to safety to prevent inadvertent criticality even though criticality events would not result in public health and safety consequences (e.g., offsite doses) that exceed guideline values.

GS = general service  
HEPA = high-efficiency particulate air (filter)  
HVAC = heating, ventilation, and air conditioning  
MCO = multi-canister overpack  
NRC = U.S. Nuclear Regulatory Commission  
SC = safety class  
SCIC = safety-class instrumentation and control  
SS = safety significant  
VPS = vacuum purge system

Important to safety categories:
- **Category A** Critical to Safe Operation  
  SSCs in this category include those whose failure or malfunction could directly result in a condition adverse to public health and safety.
- **Category B** Major Impact on Safety  
  SSCs in this category include those whose failure or malfunction could indirectly result in a condition adversely affecting collocated worker health and safety. The failure of a Category B item, in conjunction with the failure of an additional item, could result in an unsafe condition. (Note: From the definition of Category C Category B is understood to include events that could significantly damage an MCO without severe impact to collocated worker health and safety.
- **Category C** Minor Impact of Safety  
  SSCs whose failure or malfunction would not significantly reduce MCO effectiveness and would not be likely to create a situation adversely affecting public or collocated worker health and safety.
B4 2 REQUIREMENTS

This section identifies design codes, standards, regulations, and U.S. Department of Energy (DOE) orders that are required for establishing the facility safety basis. The intent is to provide only the requirements that are specific to this chapter and pertinent to the safety basis. Specific codes, standards, and requirements applicable to the CVDF are defined in HNF-SD-SNF-RD-001, *Spent Nuclear Fuel Project Standards/Requirements Identification Document*. Table B4-3 provides a summary table of general standards imposed at the CVDF for safety SSCs. Letter 99-SFD-134, *Approval of Spent Nuclear Fuel (SNF) Project Regulatory Requirement* (Loscoe 1999) provides justification for using ASME B31.3, *Process Piping Code*, with augmented quality requirements, for safety-class process piping and valves rather than using ASME *Boiler and Pressure Vessel Code*, Section III (ASME 1995).

The following DOE orders are applicable to the safety basis for the facility:

- DOE Order 6430 1A, *General Design Criteria*. This order provides general criteria and guidance for facility and system design. Division 1300-3, "Safety-Class Criteria," provides design, fabrication, and testing standards for safety-class items. Compliance with this order is documented in compliance matrices (SNF-4547), which provides a requirement-byrequirement evaluation of the compliance of the CVDF design to DOE Order 6430 1A. The contractor will maintain the design configuration, including compliance to DOE Order 6430 1A, throughout facility operation.

- DOE Order 5480 23, *Nuclear Safety Analysis Reports*. This order sets the requirements for preparing safety analysis reports, and Attachment 1 to the order includes guidance to meet the requirements. DOE-STD-3009-94 supplements DOE Order 5480 23.

- DOE Order 5480 28, *Natural Phenomena Hazards Mitigation*. This order establishes mitigation requirements for natural phenomena hazards and target probabilistic performance goals based on the facility performance category. Additional discussion of the natural phenomena hazard performance criteria is provided in Section B1.5. DOE Order 5480 28 is implemented via WHC-SD-SNF-DB-010, *Cold Vacuum Drying System Natural Phenomena Hazards*.

- DOE Order 5480 22, *Technical Safety Requirements*. This order sets the requirements for the development and preparation of a TSR document, which is prepared separately.
Table B4-3  Required Codes and Standards for the Cold Vacuum Drying Facility

<table>
<thead>
<tr>
<th>Category/application</th>
<th>Safety significant SSCs</th>
<th>Safety-class SSCs</th>
</tr>
</thead>
<tbody>
<tr>
<td>Structural including concrete and equipment skids</td>
<td>PC 2 (UBC Zone 2B (ICBO))  AISC M016  ACI 318  AWS D1 1</td>
<td>PC 3  ANSI N690  ACI 349</td>
</tr>
<tr>
<td></td>
<td>Guidance on applicable codes and standards is also provided by performance category in DOE STD 1021 93</td>
<td></td>
</tr>
<tr>
<td>Equipment anchorage</td>
<td>PC 2 (DOE STD 1020 94)</td>
<td>PC 3 (DOE STD 1020 94)</td>
</tr>
<tr>
<td>Process equipment vessels and tanks</td>
<td>ASME Code Section VIII  AWWA D100  UL 58  UL 142</td>
<td>ASME Code Section III*</td>
</tr>
<tr>
<td>Process equipment piping and valves</td>
<td>ANSI/ASME B31 1  B31 3  B31 5  B31 9  B 16 series</td>
<td>ASME B31 3*</td>
</tr>
<tr>
<td>Process equipment pumps and compressors</td>
<td>ANSI/ASME B73 1M  B73 2M  ASME Code Section VIII</td>
<td>ASME Code Section III</td>
</tr>
<tr>
<td>Process equipment heat exchangers</td>
<td>ASME VIII  TEMA</td>
<td>ASME Code Section III*</td>
</tr>
<tr>
<td>Process hood</td>
<td>ASHRAE  ACGIH</td>
<td>Same as SS</td>
</tr>
<tr>
<td>Process equipment prefilters and HEPA filters and HVAC ducting</td>
<td>ASHRAE 52 68  SMACNA  ANSI/ASME N509  N510  ERDA 76 21  DOE STD 3020 97  ASME AG 1</td>
<td>Same as SS</td>
</tr>
<tr>
<td>Mechanical handling cranes</td>
<td>CMAA 74  ANSI/ASME B30 11  B30 16  B30 17</td>
<td>CMAA nuclear sections</td>
</tr>
<tr>
<td>Electrical</td>
<td>NFPA 70  IES RP1  IES RP7</td>
<td>IEEE 308  IEEE 338  IEEE 344  IEEE 379  IEEE 384  IEEE-603 (Section 5 8)</td>
</tr>
<tr>
<td>Instruments and controls</td>
<td>ISA S5 1  ISA S5 4  ISA S18 1  ISA S20  NUREG 0700 and 0800</td>
<td>IEEE 344  IEEE 379  IEEE 384  IEEE 603  IEEE-627  NEMA ICS 6</td>
</tr>
<tr>
<td>Facility fire protection – hydrogen</td>
<td>NFPA 69  NFPA 70</td>
<td>Same as SS</td>
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<tr>
<td>Industrial hygiene hazards</td>
<td>OSHA (29 CFR 1910)</td>
<td>Same as SS</td>
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<tr>
<td>Health physics</td>
<td>10 CFR 830  ANSI HPSSC 6 8 1  ANSI/ANSI 57 7  ANSI N13 6  N13 15  N42 17B and C  N323  NCRP 57 and 58</td>
<td>Same as SS</td>
</tr>
</tbody>
</table>

*The Spent Nuclear Fuel Project has justified the use of ASME B31 3 Process Piping Code with augmented quality assurance requirements for the Cold Vacuum Drying Facility safety-class piping and valves (Loscoe 1999)

HVAC = heating ventilation and air conditioning
HEPA = high-efficiency particulate air (filter)
PC = performance category
SS = safety significant
SC = safety class
SSC = structure system or component
UBC = Uniform Building Code
DOE-STD-1020-94 *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities* This standard provides the methodology for design of the CVDF for natural phenomena hazard events and is implemented via WHC-SD-SNF-DB-010

In Letter 95-SFD-167, *Implementation of K Basins Spent Nuclear Fuel Project (SNFP) Regulatory Policy* (Sellers 1995), DOE established the requirement that new SNF Project facilities such as the CVDF achieve "nuclear safety equivalency" to comparable U S Nuclear Regulatory Commission (NRC)-licensed facilities. The SNF Project identified the NRC requirements that were needed in addition to existing and applicable DOE requirements to establish nuclear safety equivalency. These NRC requirements and the process used to identify them are documented in HNF-SD-SNF-DB-003

Applying NRC equivalency requires that SSCs important to safety be identified and a graded approach applied using the definitions in HNF-SD-SNF-DB-003. Important to safety is subdivided into the categories defined in HNF-SD-SNF-DB-003

- **Category A — Critical to Safe Operation**

  SSCs in this category include those whose failure or malfunction could directly result in a condition adverse to public health and safety. Important-to-safety SSCs in this category are classified as safety class, as defined in DOE Order 6430 1A, with the additional requirements therein.

- **Category B — Major Impact on Safety**

  SSCs in this category include those whose failure or malfunction could result in a condition adversely affecting collocated worker health and safety. Note that from the definition of Category C, Category B is understood to include events that could significantly damage the MCO without severe impact to public health and safety. SSCs in this category are classified as safety significant as defined in DOE-STD-3009-94.

- **Category C — Minor Impact on Safety**

  SSCs whose failure or malfunction would not significantly reduce containment and would not be likely to create a situation adversely affecting public or collocated worker health and safety. SSCs in this category are classified as general service (safety-class 3 [nonsafety] in DOE Order 6430 1A).

Additional discussion of the NRC criteria for mitigation of natural phenomena is provided in Section B.12.
Specific natural phenomena hazards design requirements implementing NRC equivalency requirements have been established for the CVDF in WHC-SD-SNF-DB-010

B4.3 SAFETY-CLASS SYSTEMS, STRUCTURES, AND COMPONENTS

Safety-class items are SSCs, including portions of process systems, whose failure could adversely affect the environment or the safety and health of the public. Safety-class SSCs also include the items defined in DOE Order 6430 1A, Division 1300-3. Detailed definitions of safety-class SSCs are provided in Section 3.4.1 of the SNF Project FSAR.

DOE Order 6430 1A identifies several aspects of safety-class SSCs that must be addressed in an enhanced manner. A summary discussion of the requirement and the methodology implemented by the project to impose these requirements in the design is provided below.

- Requirement

Safety-class items shall be subject to appropriately higher quality design, fabrication, and industrial test standards and codes to increase the reliability of the item and to allow credit to be taken for its capabilities in a safety analysis. Safety-class items shall be designed to the ASME Boiler and Pressure Vessel Code, Section III, Class I (ASME 1995), or to other comparable safety-related codes and standards that are appropriate for the system being designed.

- Implementation

HNF-SD-SNF-DRD-002, Cold Vacuum Drying Facility Design Requirements, identifies specific and additional requirements and reference standards and guides to be imposed on the design of safety-class SSCs. These are identified in Tables 5-2 and 6-2 of HNF-SD-SNF-DRD-002 and are listed in Table B4-3. Safety-class items were procured or fabricated to ASME NQA-1, Quality Assurance Program Requirements for Nuclear Facilities, requirements of the SNF Project quality assurance program plan (HNF-SD-SNF-QAPP-004). Safety-class piping was designed and fabricated in accordance with ASME B31.3 with augmented quality assurance requirements. An evaluation was performed to demonstrate that this is a comparable and appropriate code with the enhanced quality requirements (Loscoe 1999).

Single-Failure Criterion and Redundancy

- Requirement

The design shall ensure that a single failure (as defined in DOE Order 6430 1A) does not result in the loss of capability of a safety-class system to accomplish its required...
safety functions. To protect against single failures, the design shall include appropriate redundancy and shall consider diversity to minimize the possibility of concurrent common-mode failures of redundant items.

- Implementation

Safety-class systems have been designed to ensure that a single failure does not result in loss of capability to accomplish its safety function. Analysis to demonstrate final design compliance with IEEE 603-1991, *Criteria for Safety Systems for Nuclear Power Generating Stations*, single-failure criteria was required by HNF-SD-SNF-DRD-002. Single-failure analysis was conducted for all safety-class systems and component functions. This is documented in SNF-4290 *Analysis of Failure Modes and Their Effects on Safety-Class Systems and Components of the Cold Vacuum Drying Facility*. Safety-class SSCs meet the single-failure criterion of DOE Order 6430 1A, as shown in SNF-4290.

**Equipment Environment Considerations**

- Requirement

Safety-class items shall be designed to withstand the effects of, and be compatible with, the environmental conditions associated with operation, maintenance, shutdown, testing, and accidents. The environmental capability of equipment shall be demonstrated by appropriate testing, analysis, and operating experience, or other methods that can be supported by auditable documentation or by a combination of these methods. Equipment qualification shall provide assurance that safety-class items will be capable of performing required safety functions under DBA conditions. The qualification shall demonstrate that the equipment can at least perform for the period of time that its safety functions are required. Environmental temperature, pressure, and humidity shall be based on the most severe postulated accident affecting the particular item.

- Implementation

All safety components were designed to function under worst-case internal and external environmental conditions. These conditions are composed of design basis events and process upset conditions identified in Section 3.5.2 of HNF-SD-SNF-DRD-002, and the associated internal and external environmental conditions listed. Safety detection or protection are provided where the design could not guard against these impacts. The design mitigates conditions that represent a potential common-mode failure, as demonstrated in single-failure analyses.
The CVDF SSCs and their safety function and design criteria, codes and standards, and quality assurance requirements that are required to establish the safety basis of the SSCs are provided in HNF-SD-SNF-SEL-002, *Spent Nuclear Fuel Project Cold Vacuum Drying Facility Safety Equipment List* Environmental qualification of equipment is an important aspect of the safety equipment list. It is important to note that two ranges of environmental conditions have been identified for procurement and performance of the SSCs as follows.

**Condition A**  
60°F at 40% relative humidity (RH), 75°F at 25% RH

**Condition B**  
40°F at 60% RH, 115°F at 22% RH

(Note: The CVDF is a mild environment as defined in IEEE-323-1983 similar to condition A above)

Safety-class SSCs meet the more harsh requirements of condition B. In addition, the process bay temperature sensor is a control to prevent continued processing of the MCO in an environment that could exceed safety-class component temperature limits. If the temperature exceeds the parameter limit of 95°F, the bay monitoring system is designed to trip the safety-class instrumentation and control (SCIC) system relay that isolates the MCO and activates the safety-class helium (SCHe) system.

Seismic qualification criteria for the SSCs is also provided in HNF-SD-SNF-SEL-002, as follows:

**Condition A**  
Maintain critical function during and after a seismic event  
(certification to DOE-STD-1020-94, IEEE 344-1987, or equivalent)

**Condition B**  
Maintain pressure boundary before and after a seismic event  
(certification to DOE-STD-1020-94, IEEE 344-1987, or equivalent)

**Condition C**  
Maintain critical function before, during, and after a seismic event  
(certification to DOE-STD-1020-94, IEEE 344-1987, or equivalent)

**Condition 3/1**  
A general-service, safety-significant, or safety-class component cannot impact a safety-class component during a seismic event  
(certification to WHC-SD-GN-DGS-30006)

Safety-class SSCs cover the full range of seismic conditions depending on the requirements for specific components and equipment. Qualification of the safety-class and safety-significant equipment and components for the CVDF has been accomplished by vendor certification and/or testing. Shaker-table testing has been used to qualify equipment required to maintain operability after a seismic event has occurred.
Bay temperature alarm/interlock and seismic shutdown ensure that SCIC components can perform their safety functions for the period of time required.

- Requirement

Equipment operability qualification testing or a combination of testing and analysis shall be the preferred method of demonstrating the operability of fluid system components, mechanical equipment, instrumentation, and electrical equipment that are required to operate during and following a design basis earthquake (DBE).

- Implementation

The CVDF SCIC performance specification contains prescriptive testing and analysis requirements to demonstrate the operability of the SCIC system and all safety-class components that are required to operate during and after a DBE. These safety-class components are designated as performance category 3 in HNF-SD-SNF-SEL-002. An independent review of the seismic adequacy and testing of the performance category 3 components (safety-class components that are relied upon during and after an earthquake) has been performed by qualified personnel.

Maintenance

- Requirement

The design shall consider the maintainability factors unique to the specific equipment to be used in the facility. Facility design shall provide for routine maintenance, repair, or replacement of equipment subject to failure.

- Implementation

The maintenance philosophy for the CVDF with its short operating life is that if any equipment does fail, it will be replaced rather than repaired. Containment glove-bags will be used for breaking potentially contaminated lines to accomplish confinement of radioactive materials during maintenance. No contaminated maintenance or repair capability is required or provided at the CVDF.

- Requirement

Safety-class items shall be designed to allow inspection, maintenance, and testing to ensure their continued function, readiness for operation, and accuracy.
Implementation

The designs of safety-class systems provide for inspection, testing, and maintenance. Safety-class valves, which include local position indication, are of three-piece construction that provides ease of maintenance and allows for in-place pressure testing at system pressures. Safety-class instrumentation is designed with redundancy test functions, and provisions for calibration. A critical point in the maintenance is that the facility has an anticipated life of two to five years and few failures will occur during that period of operation.

Requirement

The design of all process equipment shall include features to minimize self-contamination of the equipment, piping, and confinement areas. The design of process equipment also shall include features to minimize the spread of contamination out of local areas.

Implementation

The MCO HEPA filter will minimize contamination of the vacuum purge system (VPS) piping, process vent piping, and SCHe vent path piping. Other safety-class components fail safe in a closed position which minimizes spread of contamination. Electrical equipment is not capable of self-contamination.

Testing

Requirement

The design shall include provisions for periodic testing of monitoring, surveillance, and alarm systems. In addition, the design shall provide the capability to test periodically, under simulated emergency conditions, safety-class items that are required to function under emergency conditions.

Implementation

HNF-SD-SNF-DRD-002, Rev 2, Section 5 1 3 describes the testing, surveillance, and maintenance requirements for all equipment, including general-service and safety-class equipment and systems. These requirements will be accomplished at a sufficient level of testing capability. Process upset conditions can be simulated for periodic testing. Trip and test circuits are also provided for system testing.
Annex B — Cold Vacuum Drying Facility

- **Requirement**

  All safety-class systems for which credit is taken in the accident analysis shall be in-place testable in terms of pressure filtration or removal efficiency, alarm capability, leak resistance, and the like. Safety-class items shall be designed to be tested on a regular schedule.

- **Implementation**

  There are no safety-class ventilation features at the CVDF, therefore, ventilation stack monitoring, and HEPA filtration system testability is not applicable, although the capability is provided. Safety-class systems provided to prevent accidents that have the potential for releases that exceed the offsite accidental release criteria established for the SNF Project are tested on regular schedules established in the TSRs.

  The enhanced requirements for safety-class systems identified in DOE Order 6430 1A are specifically imposed on the CVDF design of safety-class SSCs by Section 5.1 of HNF-SD-SNF-DRD-002. Section 5.1 specifically addresses safety functional requirements, testing, surveillance, and maintenance requirements, environmental qualification, seismic qualification, and human factors. Single-failure analysis requirements are imposed for specific systems. Compliance to these requirements is documented in the DOE Compliance Matrices (Williams 1999).

  Safety-class function can be accomplished with safety-class components in safety-significant or general-service systems. Failure of general-service portions of the system results in a failure mode in which safety-class functions of the safety-class components are not required (e.g., a broken general-service helium line results in a condition where the safety-class flow meters are no longer required). The SSCs credited with a safety-class function in Chapter B3.0 and detailed in this section are identified below:

  - SCIC system
  - SCHe system
  - Cask-MCO safety-class components
  - Tempered water (annulus) system safety-class components
  - VPS safety-class components
  - General-service helium system safety-class components
  - Process water conditioning (PWC) system safety-class components

  The safety-class SSCs for the CVDF are described in Sections B4.3.1 through B4.3.7, which provide information about the safety function and the suitability of the safety analysis inputs and assumptions. These SSCs have been identified in Chapters B3.0 and B6.0 as safety class because they prevent or mitigate the potential offsite consequences of the postulated accidents or because they provide features that prevent inadvertent criticality.
Nuclear safety equivalency to comparable NRC-licensed facilities is implemented by compliance with existing and applicable DOE requirements coupled with the additional requirement items documented in HNF-SD-SNF-DB-003 that are applicable to the CVDF. Compliance with the additional HNF-SD-SNF-DB-003 requirements is documented in a compliance matrix (Williams 1999) and summarized in Table B4-4.

**B4 3 1 Safety-Class Instrumentation and Control System**

**B4 3 1 1 Safety Function** The overall safety function of the SCIC system is to protect the MCO from hazardous conditions during process upsets. The SCIC system also responds to key abnormal process parameters by providing signals to isolate the MCO and actuate the SCHe system and/or remove power to the tempered water (annulus) system heater. The SCIC system also responds to signals from the seismic detection system and the process bay high-temperature sensors by isolating the MCO and actuating the SCHe system. These responses are intended to cause actions that will prevent or mitigate a wide range of potential accidents, including the MCO overpressurization, thermal runaway reaction, and internal and external hydrogen explosion accident scenarios described in Chapter B3 0.

The SCIC system performs the following safety functions for these accidents:

- **MCO overpressurization accident** — Process parameter upsets characteristic of an MCO overpressurization are detected by the SCIC system, which responds by isolating the MCO from the process systems and actuating the SCHe system, which vents the MCO. A seismic event and/or a high process bay temperature initiates an SCIC system trip that isolates the MCO and actuates the SCHe system, thereby pressurizing and purging the MCO. The tempered water heater will be de-energized by the SCIC system. An annulus water high temperature trip will de-energize the tempered water heater. The SCIC system alarms in the control room on detection of a low annulus water level and initiates operator response.

- **Thermal runaway reaction** — The SCIC system responds to process parameter upsets (e.g., high annulus water inlet temperature, exceeding time under vacuum, and low annulus water level) that are indicative of upset conditions that may lead to a thermal runaway accident. For some process upsets, the SCIC system provides signals to isolate the MCO from the process systems and actuate the SCHe system to pressurize and purge the MCO. A seismic event or high temperature in a process bay will initiate SCIC system trips that actuate MCO isolation and activate the SCHe system. The high annulus water temperature trip will de-energize the tempered water heater. The SCIC system alarms on a low annulus water level in the control room, which requires operator action in the bay to stop the leak and to restore water to the annulus.
### Table B4-4  Summary of Compliance with Additional U S  Nuclear Regulatory Commission Equivalency Items  (3 sheets)

<table>
<thead>
<tr>
<th>Item No</th>
<th>Requirement Area</th>
<th>Compliance Summary</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Fire Protection</td>
<td>The fire hazard analysis report (SNF-4268) evaluates the CVDF design and compares the requirements of DOE Orders 5480 7A and 6430 1A to NRC requirements. DOE requirements provide adequate fire protection to achieve nuclear safety equivalency. A review of 10 CFR 72 licensed facilities did not identify any additional fire protection requirements.</td>
</tr>
<tr>
<td>2</td>
<td>Natural Phenomena</td>
<td>The SNF Project's seismic and other natural phenomena criteria (SAR Sections B1 4 and B1 5) achieves NRC equivalence in safety. Loss of shared utilities is evaluated in HNF SD SNF HIE 004 Rev 4 CVDF Hazard Analysis Report to not be a safety concern for CVDF.</td>
</tr>
<tr>
<td>3</td>
<td>Electrical equipment qualification</td>
<td>CVDF design, as discussed in SAR Section B4 3 does not result in any harsh environment conditions. Qualification for mild environments is performed in accordance with IEEE 627 1980.</td>
</tr>
<tr>
<td>4</td>
<td>Loss of AC power</td>
<td>Loss of AC power was evaluated by the CVDF hazard analysis (HNF SD SNF HIE-004) and this SAR (Sections B2 8 5 B3 4 2 7 2 and B4 4 7) Safety significant standby diesel power is provided as required for certain accident mitigation functions.</td>
</tr>
<tr>
<td>5</td>
<td>Lead storage battery power</td>
<td>Not applicable to CVDF</td>
</tr>
<tr>
<td>6</td>
<td>Lead storage battery power</td>
<td>Not applicable to CVDF</td>
</tr>
<tr>
<td>7</td>
<td>I&amp;C system design (IEEE 603 1991)</td>
<td>Safety class I&amp;C as defined in SAR Table B4 2 meets IEEE 603 1991. Single failure analysis was performed for safety-class structures system, and components using IEEE 379 1994 methodology (SNF 4290).</td>
</tr>
<tr>
<td>8</td>
<td>Nuclear criticality alarms per ANSI/ANS 8 3</td>
<td>Not applicable to CVDF as discussed in SAR Section B6 6. Criticality detection and alarms are not required per DOE Order 5480 24 paragraph 70(3).</td>
</tr>
<tr>
<td>9</td>
<td>Human factors</td>
<td>Applicable NRC requirements and guidance were considered as part of the human factors evaluation and design performed for the CVDF and presented in SAR Chapter 13.</td>
</tr>
<tr>
<td>10</td>
<td>Code class determination</td>
<td>Applicable NRC guidance was considered (e.g., NRC Regulatory Guide 1 26) in assigning the applicable code class as discussed in SAR Section B4 3. ASME B31 3 with supplemental requirements was selected as the appropriate piping code for the CVDF application.</td>
</tr>
<tr>
<td>11</td>
<td>ASME Section III (ASME 1995) code cases</td>
<td>Not applicable to CVDF</td>
</tr>
<tr>
<td>12</td>
<td>HEPA filtration design and testing</td>
<td>HEPA equivalency is provided by application of ANSI/ASME N509 1989 and ANSI/ASME N510 1989 for HEPA filter design and testing as discussed in SAR Sections B4 4 2 2 and B4 4 3 2.</td>
</tr>
<tr>
<td>13</td>
<td>Light water reactor fuel handling systems and spent fuel storage facility design requirements</td>
<td>CVDF cranes are classified general service as discussed in SAR Section B2 9 3. A detailed evaluation of CVDF design compliance with the guidance of the applicable sections of the standard used for NRC licensed fuel handling and spent fuel storage facilities (ANSI/ANS 57 1 and 57 2) was performed in order to establish that NRC equivalency was achieved as appropriate.</td>
</tr>
</tbody>
</table>
# Table B4-4: Summary of Compliance with Additional U.S. Nuclear Regulatory Commission Equivalency Items (3 sheets)

<table>
<thead>
<tr>
<th>Item No</th>
<th>Requirement Area</th>
<th>Compliance Summary</th>
</tr>
</thead>
<tbody>
<tr>
<td>14</td>
<td>Lessons learned from NRC Generic Letters 88-14, 88-10 and 89-13</td>
<td>Lessons learned from NRC Generic Letter 88-14 (instrument air supplies) were incorporated in the CVDF design as discussed in SAR Sections B4-324, B4-3-54, B4-3-64, and B4-3-74. Generic Letters 89-10 and 89-13 are not applicable to CVDF.</td>
</tr>
<tr>
<td>15</td>
<td>Reporting of defects</td>
<td>Defect and noncompliance reporting is implemented for CVDF in a fashion comparable to 10 CFR 21 as discussed in SAR Section 14.2.</td>
</tr>
<tr>
<td>16</td>
<td>QA Program approval</td>
<td>DOE approval is required for QA programs and program changes. The DOE process, while not identical to NRC regulation, is judged to achieve objectives equivalent to NRC regulations (i.e., 10 CFR 50.54(a)).</td>
</tr>
<tr>
<td>17</td>
<td>Occurrence reporting</td>
<td>Occurrence reporting consistent with DOE O 2321A is implemented for CVDF during design and construction and is considered to be consistent with the comparable NRC regulation (i.e., 10 CFR 50.55(e)).</td>
</tr>
<tr>
<td>18</td>
<td>QA Program application</td>
<td>SAR Section B14.2 commits the CVDF to continued application of the QA Program. CVDF QA program meets ASME NQA 1. The ASME NQA 1 standard is an acceptable method for meeting 10 CFR Appendix B or 10 CFR 72, Subpart G.</td>
</tr>
<tr>
<td>19</td>
<td>Counterfeit/defective equipment</td>
<td>The SNF Project monitors industry experience (including NRC notices, bulletins, and circulars) for counterfeit and defective equipment as discussed in SAR Section 14.2.</td>
</tr>
<tr>
<td>20</td>
<td>Radiation controls</td>
<td>No high radiation areas are anticipated in CVDF and public dose criteria will not be exceeded for normal operation and anticipated occurrences as discussed in SAR Section B7.6.</td>
</tr>
<tr>
<td>21</td>
<td>Radiation exposure</td>
<td>Estimated dose-equivalent exposures for CVDF are well below both DOE and NRC criteria as discussed in SAR Section B7.6.</td>
</tr>
<tr>
<td>22</td>
<td>Deleted (see item 29)</td>
<td>Deleted</td>
</tr>
<tr>
<td>23</td>
<td>ALARA</td>
<td>NRC Regulatory Guide 8.8 was used for CVDF design and ALARA programs as discussed in SAR Section B7.4.</td>
</tr>
<tr>
<td>24</td>
<td>Safety analysis report format and content</td>
<td>Inclusion of the appropriate format and content information is assured by use of the check list from HNF SD SNF SP-012. Table 5. The CVDF has a short term mission (design life of 5 years) and low dose impacts on workers and the public.</td>
</tr>
<tr>
<td>25</td>
<td>Effluent monitoring</td>
<td>The CVDF stack monitoring system provides the gaseous effluent monitoring information required by 10 CFR 70.59 and 10 CFR 50.36a. There are no liquid effluent releases from the CVDF because the liquids are retained in storage tanks and basins and transferred elsewhere for disposal.</td>
</tr>
<tr>
<td>26</td>
<td>General design criteria</td>
<td>A review of applicable sections of the general design criteria (10 CFR 50 Appendix A) concluded that NRC nuclear safety equivalency is achieved by application of DOE Order 6430.1A (attachment to 99 SNF/CSH 005 19 July 1999). The criteria specific to nuclear reactor systems are not applicable to CVDF.</td>
</tr>
</tbody>
</table>
Table B4-4 Summary of Compliance with Additional US Nuclear Regulatory Commission Equivalency Items (3 sheets)

<table>
<thead>
<tr>
<th>Item No</th>
<th>Requirement Area</th>
<th>Compliance Summary</th>
</tr>
</thead>
<tbody>
<tr>
<td>27</td>
<td>Criticality safety value for $k_{eff}$</td>
<td>A requirement to maintain a $k_{eff}$ value of $\leq 0.95$ is imposed for the multi-cask overpack and the CVDF design, as discussed in SAR Section B6 1</td>
</tr>
<tr>
<td>28</td>
<td>Fuel storage facility requirements</td>
<td>Not applicable to CVDF</td>
</tr>
<tr>
<td>29</td>
<td>Safety classification</td>
<td>CVDF implementation of NRC 'important to safety' classification is documented in SAR Section B4 2 and Table B4 2</td>
</tr>
</tbody>
</table>

ALARA = as low as reasonably achievable  
CVDF = Cold Vacuum Drying Facility  
DOE = US Department of Energy  
FSAR = final safety analysis report  
HEPA = high-efficiency particulate air (filter)  
I&C = instrumentation and control  
NRC = US Nuclear Regulatory Commission  
QA = quality assurance  
SNF = spent nuclear fuel
Internal hydrogen explosion — The SCIC system responds to process parameter upsets that could be indicative of excessive hydrogen inside the MCO or a line break that would allow the ingress of air. If selected process parameters are found to be outside limits, the SCIC system actuates the SCHe system, which is designed to maintain the hydrogen and oxygen concentration inside the MCO below the lower flammability limit. The SCIC system also detects high process bay temperatures and seismic events that could impair the system and responds by isolating the MCO and activating the SCHe system. A high water temperature in the tempered water (annulus) system results in shutting off power to the tempered water (annulus) system heater. The tempered water (annulus) low water level is monitored to sound an alarm for operator response.

External hydrogen explosion — The SCIC process bay high temperature detection is credited to protect assumptions of the fire hazards analysis for preventing damage to safety-class equipment components in the process bay. TSR response to an ISO & PURGE alarm requires that process bay temperature be verified and high temperature returned to the acceptable range to protect the temperature assumptions used in with the fire hazards analysis combustible loading limits in the process bays.

The SCIC system is designated safety-class because of its role in preventing the occurrence of the MCO thermal runaway and MCO overpressurization accidents, which, unmitigated, would exceed offsite release limits. The SCIC system also plays a role in preventing the internal hydrogen explosion accident, which has unmitigated consequences that exceed onsite risk acceptance guidelines but do not exceed offsite release limits. The SCIC system performs a safety-significant function for prevention of this accident.

**B4 3 1 2 System Description** The SCIC system is a dedicated safety-class system whose function is to aid in the prevention of the thermal runaway, MCO overpressurization, and MCO internal explosion events by responding to abnormal conditions and actuating one or more safety-class systems or alarms. The SCIC system monitors specific cold vacuum drying process parameters for each process step, seismic condition, and process bay temperature. The system responds to anomalies that, if undetected, could result in an accident condition.

Two SCIC panels (instrument racks) are located in each process bay. One of the two panels contains a programmable logic controller (PLC) that provides one of the two logic trains for that bay. The other PLC for that bay is located in the adjacent bay. Figure B2-27 shows the layout of the SCIC system in each bay and in the control room. Redundant instruments and control functions are provided to the two SCIC logic trains, A and B. The PLC performs the various setpoint comparisons and provides safety-related timing functions (see Figure B2-26 for a summary of the SCIC system functional circuitry). Included in this logic is a vacuum limit timer that monitors the time under vacuum to limit heatup of the fuel. Vacuum conditions reduce thermal conduction from the MCO. After the vacuum cycle, a minimum time at or slightly above atmospheric pressure is required to cool down the fuel. The SCIC system ensures that exceeding the time under vacuum, or returning to vacuum without first meeting the minimum time above...
atmospheric pressure, will result in an MCO ISO & PURGE trip (i.e., the MCO is isolated by closure of the VPS, general-service helium system, PWC system, and deionized water isolation valves and the SCHe system is activated).

Physical and electrical separation is achieved in accordance with Institute of Electrical and Electronics Engineers requirements. Each process bay has instrument racks with SCIC instrument panels for monitoring and control of trains A and B for that process bay. The panels provide the following capabilities in each bay:

- A local manual ISO & PURGE button that isolates the MCO and activates the SCHe system. Activation of either the train A or the train B ISO & PURGE push button will initiate a trip that will close one set of safety-class MCO isolation valves to sufficiently isolate the MCO and will activate the SCHe system. Both ISO & PURGE buttons must be activated to provide MCO isolation by both safety-class valves in each process system line.

- ISO & PURGE trip reset button

- Tempered water high temperature trip reset button

- Trip status and indicator lights for bay ISO & PURGE trip, tempered water high temperature trip, seismic trip, process bay high temperature trip, and cask annulus low level alarm status

- Logic test switches

In a similar manner, the CVDF control room has SCIC system train A and B control room panels for each bay with the following capabilities:

- SCIC mode selection panel with duplicate key switches for the following operation sequence selections: (1) BYPASS, (2) HEATUP, (3) DRAIN, (4) PURGE/FLUSH, (5) DRYING, (6) PROOF, and (7) PRESSURE TEST. The two panels in the control room have key switches for all operations in process bays 2, 3, 4, and 5.

- An annunciator and control panel with large lighted alarms for BAY ISO & PURGE, ANNULUS LOW LEVEL and PWC LOW FLOW, alarm test capability, alarm acknowledgment, and annulus low level alarm bypass switches, and ISO & PURGE buttons for MCO isolation and activation of the SCHe system for each of the process bays.

During cold vacuum drying of an MCO, the SCIC system is placed into the mode appropriate for each processing stage, as shown in Figure B2-11. When the MCO is first received, the SCIC system is in the BYPASS mode while the cask headspace is being vented, the cask lid is being removed, the process hood seal ring assembly is being bolted on, and the two
sealing bladders are being inflated. The SCIC system is switched to the HEATUP mode after the port connectors are attached during heatup the MCO headspace is vented. When water is to be removed from the MCO, the SCIC system switch is placed in the DRAIN mode. After MCO draining is complete, the SCIC system switch is positioned in the PURGE/FLUSH mode to flush particulate from the drain line. When the MCO is to be evacuated for vacuum drying, the SCIC system switch is placed in the DRYING mode, and when the drying is completed and proof verification is to be performed, the mode switch is placed in the PROOF position. After proof checking, the SCIC system switch is placed in the PRESSURE TEST mode for the final 1-hour pressure rebound test. As a final step, the MCO port valves are closed and the MCO is prepared for shipment. During this period, the SCIC system is placed in the BYPASS mode. Each of these modes enables the SCIC system to monitor setpoints from the different sensors and make available appropriate logic circuitry for each specific processing step. Sensors located in the process systems that are connected to the SCIC system are shown in Figure B4-1. These sensors provide the following input to the SCIC system:

- MCO pressure, from the VPS (pressure transmitters with a range of 0 to 100 torr and pressure transmitters with a range of -14.7 lb/in² gauge to +12 lb/in² gauge)
- Helium purge flow rate, from the general-service helium system (flow-indicating transmitters)
- Tempered water high temperature switch, from the tempered water (annulus) system
- Cask-MCO annulus water low-level switch from the tempered water (annulus) system
- Process bay temperature switch
- Seismic recorder and trip status, integral to the SCIC system

The seismic monitors will activate the SCIC system trip independent of the logic circuits (see Figure B2-26). The seismic monitors directly trip to the SCIC system output relays, which actuate the SCH system and MCO isolation valves. A seismic event will also trip the tempered water high temperature protection circuit that shuts down the tempered water heater. In addition, the seismic trip will shut down power to the PWC system circulation pumps. Loss of electrical power will de-energize the SCIC system output relay, which will isolate the MCO, actuate the SCH system and eliminate power to the tempered water (annulus) system heater and the PWC system circulation pumps.

The SCIC system also provides safety-class alarms in the control room. The SCIC system monitors the tempered water (annulus) system low-level switch and activates an alarm that notifies the operators when the water level in the cask-MCO annulus is low. Refer to SNF-3091, SNF Project Cold Vacuum Drying Facility Safety-Class Instrument and Control System Design Description for more detailed information.
B4 3 1 3 Functional Requirements  The functional requirements needed for the SCIC system to perform its safety functions are as follows:

- **Operations and Mode Selection** — To be operational before the VPS connectors are attached to the MCO ports and then placed in the proper mode as each process step is performed.

- **Monitoring and Logic** — Monitor the input signals from the VPS pressure-transmitters and general-service helium safety-class flow meters, and perform logic and timing functions necessary to ensure that hydrogen and/or oxygen is not accumulating in the MCO. If logic or timing function parameters indicate conditions wherein hydrogen and/or oxygen could be accumulating beyond acceptable limits, the SCIC system must be capable of automatically initiating an ISO & PURGE trip. This action will bring the MCO to slightly above atmospheric pressure, prevent further ingress of air, and provide the MCO internal heat transfer medium (helium) for enhanced heat transfer to the tempered water (annulus) system.

- **Annulus Low Water Level Alarm** — Monitor the tempered water (annulus) system annulus water low level switch and provide an alarm to the operators if the level falls below the lower limit.

- **Tempered Water High Temperature Trip** — Monitor the tempered water (annulus) system annulus inlet water temperature switch and de-energize electrical power to the tempered water (annulus) system heater, if the temperature is above the setpoint limit.

- **Process Upset Conditions** — Automatically initiate an ISO & PURGE trip under process upset conditions, including loss of power.

- **Control Room Alarms** — Provide safety-class alarms for BAY ISO & PURGE ANNULUS LOW LEVEL, and the non-safety PWC LOW FLOW alarm to the CVDF control room.

- **Seismic Safe Shutdown** — To be capable of receiving signals from the seismic recorder and trip component, and, upon ground motion acceleration above the setpoint, automatically activating MCO isolation and SCHe system activation. In addition, provide shutdown of power to the tempered water (annulus) system heater.

- **High Bay Temperature** — Monitor the process bay temperature. Initiate SCIC trip, isolate the MCO, and purge via the SCHe system if the bay air temperature exceeds the setpoint.
**Nonqualified Signal Protection** — Provide protection from any nonsafety operational controls (e.g., coming from the monitoring and control system [MCS]) that could affect or degrade the safety-class operation of the SCIC system.

**Sensor Error** — Account for sensor error in trip setpoints so that operations are within specifications; process function monitoring does not lead to unsafe conditions and safe shutdown is accomplished as anticipated.

**Testing Functions** — Capable of providing periodic testing of the SCIC system functional operations.


**Seismic Qualification** — Meet performance category 3 seismic requirements for all process bay SCIC equipment.

### B4.3.1.4 System Evaluation
The SCIC systems are designed to be redundant and fail-safe. The SCIC systems identified are tied to key process parameter sensors, which when taken as a whole provide comprehensive and complete detection and response to the upset conditions identified in the accident analyses in Chapter B3.0. The systems also are designed such that internal system failures actuate the safety SSCs to which they are interlocked. This automatic actuation upon failure ensures that loss of monitoring of a key parameter does not degrade the safety function of the SCIC system.

The SCIC system provides automatic and independent active detection and response to process anomalies that, if unmitigated, would result in a safety-class or safety-significant event. To perform these functions, the SCIC system monitors process parameters, detects abnormal conditions, and actuates an ISO & PURGE or turns off power to the tempered water heater. Operator response is required if these tempered water (annulus) or PWC system components are de-activated. In addition, when required, the SCIC system provides safety-class alarms to the CVDF control room for those events that require operator actions. The SCIC system is independent of the MCS system, and the only operator function during normal operations is to control the position of the mode selector switch, including blowdown of the SCHe lines after MCO draining (the SCIC system operates the SCHe valves for 2 minutes to blowdown the system lines with helium from the general-service helium supply, when the SCIC system switch is in the PURGE/FLUSH mode), and to test the annunciator.

For the PWC system, the SCIC system provides an annunciator alarm (non-safety) unless the PWC system has been purged since entering the DRAIN mode. The alarm remains locked until the mode switch is placed in any other position than DRAIN. When the SCIC system is switched to the PURGE/FLUSH mode, an alarm will be activated after entering this mode unless the PWC system has been purged sufficiently with helium. There is about a 19-minute delay when...
the PURGE/FLUSH mode is entered, to allow for a deionized water rinse and an allowance for
flow responses that is taken into account by the PWC purge timing

Manual operation trip push buttons are provided to activate an MCO isolation and purge
from either the control room (safety-class annunciator and mode panel) or the respective bays.
The bay push buttons provide a qualified manual initiation button under all accident conditions,
however, there are no known reasons for the need for operator action to protect the MCO. The
control room manual trips are provided for nonspecific accidents or upset conditions that may
warrant a trip. Examples include control room evacuation, fire in the facility, or a loss of normal
process control. Although none of the manual trips are required to maintain the MCO in a safe
condition, operational procedures may warrant a trip. The CVDF control room and
the corresponding safety-class annunciator and mode switch panel within the control room are not
seismically qualified, as these functions are not required during a seismic event. The MCO is
protected by the safety-class seismic detection and trip function.

Operations and Mode Selection Limiting conditions for operation (see
Section B4 3 1 5) require that the SCIC system be operational before the process port
connections are made to the MCO. The mode selector switch enables or disables the interlocks,
provides selected mode status to the PLCs for enabling or disabling the protection logic, passes all
mode position status to the nonsafety MCS for process operations, and operates selected SCHe
valves to purge the SCHe piping. Table B4-5 summarizes which trips and interlocks are active
during the different mode switch positions.

Monitoring and Logic During receipt and prior to bulk MCO water drain, the thermal
runaway accident is not possible because the fuel is covered with water. An internal hydrogen
explosion will have no onsite or offsite consequence above the guidelines because no particulate is
available for release. Each of the PLC trips that are active in the various mode switch positions
are discussed in more detail in the following subsections (see Table B4-5)

To ensure safe conditions during the SCIC DRYING and PROOF modes, the MCO low
pressure purge trip remains active and additional PLC trips are activated as discussed below:

- MCO Low Pressure — After draining, when the SCIC system is in the DRAIN and
PURGE/FLUSH modes, an MCO low pressure purge trip is active and actuates the
ISO & PURGE if the general-service helium purge pressure falls below the
parameter limit of 0.24 lb/in² gauge (0.5 lb/in² gauge setpoint). This nominal
pressure parameter indicates that the MCO is pressurized with helium above
atmospheric pressure, which precludes air ingress under accident conditions. This is
discussed in SNF-4451, Cold Vacuum Drying (CVD) Set Point Determination. This
action ensures that a positive pressure is maintained in the MCO to preclude fuel
heating as a result of low thermal conductivity from operation at vacuum and
precludes further air ingress into the MCO in the event of a line break or other
accident scenario.
<table>
<thead>
<tr>
<th>Mode</th>
<th>PLC trips (Note 1)</th>
<th>Non PLC trips</th>
<th>Interlocks</th>
<th>Alarms</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 BYPASS</td>
<td>None</td>
<td>Seismic</td>
<td>None</td>
<td>Bay * purge TW annulus low level (Note 2)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>TW high temperature</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Local and remote ISO &amp; PURGE buttons</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>High bay temperature</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>PWC pump seismic trip*</td>
<td></td>
<td></td>
</tr>
<tr>
<td>2 HEATUP</td>
<td>MCO high pressure</td>
<td>Seismic</td>
<td>PWC-GOV 1<em>30 1</em>03 VPS-GOV 1<em>11 1</em>17 (drain and rinse valves)</td>
<td>Bay * purge TW annulus low level</td>
</tr>
<tr>
<td></td>
<td></td>
<td>TW high temperature</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Local and remote ISO &amp; PURGE buttons</td>
<td></td>
<td></td>
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<tr>
<td></td>
<td></td>
<td>High bay temperature</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>PWC pump seismic trip*</td>
<td></td>
<td></td>
</tr>
<tr>
<td>3 DRAIN</td>
<td>MCO high and low pressure</td>
<td>Seismic</td>
<td>None</td>
<td>PWC low flow* Bay * purge TW annulus low level</td>
</tr>
<tr>
<td></td>
<td></td>
<td>TW high temperature</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Local and remote ISO &amp; PURGE buttons</td>
<td></td>
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<td></td>
<td></td>
<td>High bay temperature</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>PWC pump seismic trip*</td>
<td></td>
<td></td>
</tr>
<tr>
<td>4 PURGE/FLUSH</td>
<td>MCO high and low pressure</td>
<td>Seismic</td>
<td>Note 3</td>
<td>PWC low flow* Bay * purge TW annulus low level</td>
</tr>
<tr>
<td></td>
<td></td>
<td>TW high temperature</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Local and remote ISO &amp; PURGE buttons</td>
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<td>High bay temperature</td>
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<td></td>
<td></td>
<td>PWC pump seismic trip*</td>
<td></td>
<td></td>
</tr>
<tr>
<td>5 DRYING</td>
<td>MCO high and low pressure</td>
<td>Seismic</td>
<td>PWC-GOV 1<em>30 1</em>03 VPS-GOV 1<em>11 1</em>17 (drain and rinse valves)</td>
<td>Bay * purge TW annulus low level</td>
</tr>
<tr>
<td>MCO prepurge</td>
<td></td>
<td>TW high temperature</td>
<td></td>
<td></td>
</tr>
<tr>
<td>MCO low purge flow</td>
<td></td>
<td>Local and remote ISO &amp; PURGE buttons</td>
<td></td>
<td></td>
</tr>
<tr>
<td>MCO 12 torr purge bypass</td>
<td></td>
<td>High bay temperature</td>
<td></td>
<td></td>
</tr>
<tr>
<td>MCO low purge flow bypass</td>
<td></td>
<td>PWC pump seismic trip*</td>
<td></td>
<td></td>
</tr>
<tr>
<td>MCO pressure decay fail</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>MCO pressure rise fail</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>MCO 8 hr vacuum limit timer</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>MCO 4 hr vacuum limit timer</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>6 PROOF</td>
<td>MCO high and low pressure</td>
<td>Seismic</td>
<td>PWC-GOV 1<em>30 1</em>03 VPS-GOV 1<em>11 1</em>17 (drain and rinse valves)</td>
<td>Bay * purge TW annulus low level</td>
</tr>
<tr>
<td>MCO prepurge</td>
<td></td>
<td>TW high temperature</td>
<td></td>
<td></td>
</tr>
<tr>
<td>MCO low purge flow</td>
<td></td>
<td>Local and remote ISO &amp; PURGE buttons</td>
<td></td>
<td></td>
</tr>
<tr>
<td>MCO 12 torr purge bypass</td>
<td></td>
<td>High bay temperature</td>
<td></td>
<td></td>
</tr>
<tr>
<td>MCO low purge flow bypass</td>
<td></td>
<td>PWC pump seismic trip*</td>
<td></td>
<td></td>
</tr>
<tr>
<td>MCO pressure decay fail</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>MCO pressure rise fail</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Table B4-5  Safety-Class Instrumentation and Control Trip, Interlock, and Alarm Summary  (2 sheets)

<table>
<thead>
<tr>
<th>Mode</th>
<th>PLC trips (Note 1)</th>
<th>Non PLC trips</th>
<th>Interlocks</th>
<th>Alarms</th>
</tr>
</thead>
<tbody>
<tr>
<td>7 PRESSURE TEST</td>
<td>MCO pressure rise fail</td>
<td>Seismic</td>
<td>VPS-GOV 1<em>05 1</em>09</td>
<td>Bay purge</td>
</tr>
<tr>
<td></td>
<td></td>
<td>TW high temperature</td>
<td>PWC-GOV 1<em>30 1</em>03</td>
<td>TW annulus low level</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Local and remote ISO &amp; PURGE buttons</td>
<td>VPS-GOV 1<em>11 1</em>17</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>High bay temperature</td>
<td>(drain and rinse valves)</td>
<td></td>
</tr>
</tbody>
</table>

*Non safety (not safety class or safety significant)*

Notes
1. All PLCs have an associated PLC fault trip built into the PLC logic
2. TW annulus low level is bypassed by operations when there is no MCO or when the annulus is drained for shipment.
3. During the PURGE/FLUSH mode the SCHε valves (SCHε-GOV 5*12 and 5*31) on the long process tube side are opened for 2 minutes (de-energized by the SCIC) to blow down any water that may have collected from the drain sequence

MCO = multi-canister overpack
PLC = programmable logic controller
PWC = process water conditioning
SCHε = safety-class helium
SCIC = safety-class instrumentation and control
TW = tempered water
MCO Prepurge — The MCO prepurge trip will detect a low purge flow rate or inadequate time of purge prior to pulling a vacuum in the MCO. The MCO low pressure trip is active and the MCO prepurge bypasses this trip only if the MCO has been prepurged with helium at a flow rate of >80 standard ft³/min for about 15 minutes. This trip ensures that the gas in the MCO is below the lower flammability limit for hydrogen prior to going to vacuum, which could introduce oxygen if an undetected air leak is present. Throughout the DRYING and PROOF modes, the MCO is cycled between pressure and vacuum, which requires hydrogen to be maintained at low concentrations when entering vacuum. This requirement is accomplished with the MCO prepurge, as indicated above.

MCO Low Purge Flow MCO Low Purge Flow Bypass, and MCO 12-torr Purge Bypass — The MCO low purge flow trip monitors the general-service helium supply flow and will trip and activate an ISO & PURGE if the general-service flow is below the parameter limit of 0.7 standard ft³/min (standard temperature defined as 68 °F) when above 12 torr and after a 5-minute delay. These trips are active in the DRYING and PROOF modes. The 0.7 standard ft³/min flow rate ensures hydrogen concentrations less than 8.7% in helium gas so that flammable conditions would be precluded, if an air ingress event were to occur.

The lowest vacuum level that can be reached with helium purge active is approximately 50 to 70 torr. The MCO low purge flow bypass allows for a 5-minute period with no helium flow, which is a process requirement so that a 12 torr vacuum can be reached. At a vacuum of 12 torr, the hydrogen is below the amount capable of supporting combustion and no helium purge is required. Helium flow is stopped, which allows the vacuum pump to bring the MCO pressure below the 12-torr parameter limit. If a leak occurs when the 5-minute timer is activated, the vacuum pump is still operational, and after 5 minutes an ISO & PURGE will be activated. The MCO 12-torr purge bypass trip logic is based on the vacuum reaching <12 torr safety parameter for 10 seconds and staying <12 torr thereafter. On return above 12 torr, there is a 2-minute delay to allow the MCO low purge to reset as flow is reestablished.

MCO Pressure Decay Fail — During the transition from pressure operation to vacuum, the MCO pressure decay fail trip is activated to ensure there are no line breaks that would allow continuous air cycling and to detect a degraded vacuum pump, both of which could allow fuel heating beyond limits. This trip also provides a method of detecting a line break in the helium line (downstream of the flowmeters) that would invalidate the safety-class indication from the flowmeters.

MCO Pressure Rise Fail — Similar to the MCO pressure rise decay trip, the MCO pressure must exceed the parameter limit of 0.24 lb/in² gauge from a starting point parameter limit of -10.8 lb/in² gauge in 5 minutes or an ISO & PURGE activation will occur. This again ensures that helium flow is provided through the MCO to...
preclude developing flammable gas mixtures. Calculations that support the use of 5 minutes for the timing period are listed in SNF-4301, *Gas Composition Transients in the Cold Vacuum Drying Facility*. The 5-minute trip will identify a defective vacuum pump that precludes operating under conditions where a degraded vacuum would be insufficient to remove hydrogen generated in the MCO.

- **MCO Vacuum Limit Timer (8- or 4-Hour)** — A vacuum limit timer trip will limit the time the MCO is under vacuum, defined as being less than 0 lb/in² gauge parameter limit. This requirement ensures internal MCO temperatures stay within parameters by limiting time under vacuum, which would reduce thermal conductivity. The 4 hours at pressure will equalize the internal MCO temperatures. A limit of 8 hours for the initial vacuum cycle and 4 hours for subsequent cycles is provided. Based on thermal analysis, the 4 hours at pressure does not cool the fuel to its pre-vacuum starting temperature, and the 4 hours under vacuum is the maximum time allowed for all subsequent cycles. The MCO must be returned to pressure for 4 hours between vacuum cycles. Calculations supporting the 8-4-4 cycle requirements are provided in SNF-4301. Exceeding the 8-4-4 cycle requirements will initiate an MCO ISO & PURGE, thereby activating the SCHe system.

When the SCIC system is in PROOF mode, all of the above logic and trips are active except for the vacuum limit timer, which is not required because there is insufficient water for a thermal runaway condition caused by being under vacuum.

Once the MCO has completed the PROOF mode, it must undergo a final 1-hour rebound test to ensure the water content in the MCO is within specifications. In this case, the SCIC system is switched to the PRESSURE TEST mode. The VPS valves are isolated (interlocked closed) to ensure no possibility of water addition at the conclusion of the final pressure rebound test. During the pressure test, the MCO pressure rise fail trip is active to guard against any possible line break or leak situation because other trips cannot be used with the VPS valves closed.

- **MCO High Pressure** — An MCO high pressure trip is active in DRAIN PURGE/FLUSH, DRYING, and PROOF modes to preclude challenging the 30 lb/in² gauge rupture disk because of an improper isolation of the MCO without a normal or SCHe vent path being available. This trip provides protection against MCO overpressurization and provides defense-in-depth against the thermal runaway reaction accident. An MCO ISO & PURGE will be activated by the SCIC system if pressure in the MCO exceeds 12 lb/in² gauge (protects the parameter limit of <25 lb/in² gauge shown in Table B4-6 and is based on limit of instrumentation). The instrument upper range is +12 lb/in² gauge.
<table>
<thead>
<tr>
<th>SCIC trip or alarm</th>
<th>Field sensor</th>
<th>Parameter limit</th>
<th>Total instrument error</th>
<th>SCIC trip setpoint*</th>
<th>Allowable value (setpoint margin) for testing and surveillance</th>
<th>Time delay</th>
<th>Averaging delay</th>
</tr>
</thead>
<tbody>
<tr>
<td>Seismic</td>
<td>ATR 5235 5336 5437 CP120 CP121</td>
<td>≤ 0.06 g</td>
<td>0.01 g</td>
<td>≤ 0.05 g</td>
<td>≤ 0.056 g (+0.0064)</td>
<td>N/A</td>
<td>N/A</td>
</tr>
<tr>
<td>TW annulus water temperature (TW high temperature)</td>
<td>TSH 1<em>28 1</em>29</td>
<td>≤ 50 °C</td>
<td>19 °C</td>
<td>≤ 481 °C</td>
<td>≤ 495 °C (+1.4)</td>
<td>None</td>
<td>None</td>
</tr>
<tr>
<td>High bay temperature</td>
<td>TSHH 1<em>38 1</em>39</td>
<td>≤ 95 °F</td>
<td>56 °F</td>
<td>≤ 89 °F</td>
<td>≤ 91.7 °F (2.77)</td>
<td>None</td>
<td>N/A</td>
</tr>
<tr>
<td>MCO prepurge</td>
<td>FIT 1<em>20 1</em>21</td>
<td>≥ 8.0 standard ft³/min</td>
<td>0.4 standard ft³/min</td>
<td>≥ 8.4 standard ft³/min</td>
<td>≥ 8.2 scfm (-0.224)</td>
<td>None</td>
<td>&lt;4 sec maximum</td>
</tr>
<tr>
<td>MCO high pressure</td>
<td>PT 1<em>36 1</em>37</td>
<td>≤ 25 lb/in² gauge</td>
<td>0.26 lb/in² gauge</td>
<td>≤ 12 lb/in² gauge</td>
<td>≥ 12.19 lb/in² (+0.192)</td>
<td>None</td>
<td>&lt;2 sec maximum</td>
</tr>
<tr>
<td>MCO low pressure</td>
<td>PT 1<em>36 1</em>37</td>
<td>≥ 0.24 lb/in² gauge</td>
<td>0.26 lb/in² gauge</td>
<td>≥ 0.5 lb/in² gauge</td>
<td>≥ 0.41 lb/in² (-0.192)</td>
<td>None</td>
<td>&lt;2 sec maximum</td>
</tr>
<tr>
<td>MCO low purge flow</td>
<td>FIT 1<em>20 1</em>21</td>
<td>≥ 0.7 standard ft³/min</td>
<td>0.4 standard ft³/min</td>
<td>≥ 1.1 standard ft³/min</td>
<td>≥ 0.9 scfm (-0.224)</td>
<td>None</td>
<td>&lt;4 sec maximum</td>
</tr>
<tr>
<td>MCO low purge flow bypass</td>
<td>N/A</td>
<td>5 minutes</td>
<td>See Note 1</td>
<td>≤5 minutes</td>
<td>See Note 1</td>
<td>N/A</td>
<td>N/A</td>
</tr>
<tr>
<td>MCO 12 torr purge bypass</td>
<td>PT 1<em>08 1</em>10</td>
<td>≤ 12 torr</td>
<td>2 torr with 2 torr system drop</td>
<td>≤ 8.0 torr</td>
<td>≤ 9.4 torr (+1.44)</td>
<td>None</td>
<td>&lt;4 sec maximum</td>
</tr>
<tr>
<td>Vacuum limit timer</td>
<td>PT 1<em>36 1</em>37</td>
<td>≥ 0 lb/in² gauge</td>
<td>0.26 lb/in² gauge</td>
<td>≥ 0.26 lb/in² gauge</td>
<td>≥ 0.07 lb/in² (-0.192)</td>
<td>None</td>
<td>&lt;4 sec maximum</td>
</tr>
<tr>
<td>Vacuum limit timer</td>
<td>N/A</td>
<td>8-4-4 hours</td>
<td>See Note 1</td>
<td>8-4-4 hours</td>
<td>See Note 1</td>
<td>None</td>
<td>N/A</td>
</tr>
<tr>
<td>SCIC trip or alarm</td>
<td>Field sensor</td>
<td>Parameter limit</td>
<td>Total instrument error</td>
<td>SCIC trip setpoint*</td>
<td>Allowable value (setpoint margin) for testing and surveillance</td>
<td>Time delay</td>
<td>Averaging delay</td>
</tr>
<tr>
<td>-------------------------</td>
<td>--------------</td>
<td>----------------------------------------</td>
<td>---------------------------------</td>
<td>---------------------</td>
<td>---------------------------------------------------------------</td>
<td>------------</td>
<td>-----------------</td>
</tr>
<tr>
<td>MCO pressure decay fail</td>
<td>PT 1<em>36 1</em>37</td>
<td>≥ 0.24 lb/in² gauge and ≤ 11.4 lb/in² gauge</td>
<td>0.26 lb/in² gauge and 0.3 lb/in² gauge</td>
<td>≥ 0.5 lb/in² gauge and ≤ 11.7 lb/in² gauge</td>
<td>≥ 0.41 lb/in² (-0.192) and ≤ 11.51 lb/in² (+0.192)</td>
<td>5 continuous seconds</td>
<td>&lt; 4 sec maximum</td>
</tr>
<tr>
<td>MCO pressure rise fail</td>
<td>PT 1<em>36 1</em>37</td>
<td>≤ 10.8 lb/in² gauge and ≥ 0.24 lb/in² gauge</td>
<td>0.3 lb/in² gauge and 0.26 lb/in² gauge</td>
<td>≤ 11.1 lb/in² gauge and ≥ 0.5 lb/in² gauge</td>
<td>≤ 10.91 lb/in² (+0.192) and ≥ 0.41 lb/in² (-0.192)</td>
<td>5 continuous seconds</td>
<td>&lt; 4 sec maximum</td>
</tr>
<tr>
<td>Bay annulus water low level alarm</td>
<td>LSL 1<em>24 1</em>25</td>
<td>≥ 10% of gauge</td>
<td>See Note 2</td>
<td>≥ 20% of gauge (50%)</td>
<td>≥ 20% of gauge</td>
<td>None</td>
<td>None</td>
</tr>
</tbody>
</table>

Notes:
1. Time functions are not prone to errors in setpoint, therefore no error is applied. The parameter and the SCIC setpoint are equal.
2. Bay annulus water level gauge and level switch are at least 1 ft above the level of the fuel and therefore the level setpoint is well above the required safety limit.

*SNN-4451 1999 Cold Vacuum Drying (CVD) Set Point Determination Rev 0 Fluor Daniel Hanford, Incorporated, Richland Washington

MCO = multi-canister overpack
N/A = not applicable
SCIC = safety-class instrumentation and control
standard ft³/min = 68 °F and 1 atm
TW = tempered water
8-4-4 = 8 hour initial vacuum cycle 4 hour subsequent vacuum cycles 4 hour return to pressure between vacuum cycles
Annulus Low Water Level Alarm  The tempered water (annulus) system low level switch provides protection from a thermal runaway reaction by alerting the control room operators to situations where an annulus water leak has occurred. The low level switch trips the CVDF control room alarm and operators must acknowledge the alarm and initiate response.

Tempered Water High Temperature Trip  The tempered water high temperature switch trip provides protection from a thermal runaway reaction by limiting the heat from the tempered water (annulus) system. This is done by tripping the tempered water heater if the tempered water exceeds the 50 °C parameter limit. The temperature limit is based on the parameters used for the thermal calculation for the thermal runaway accident (see Section B3.4.2.5). The SCIC system precludes adding excessive heat into the MCO to protect the assumptions of the thermal calculation, which used 50 °C as the upper water temperature value. Thermal inertia and water flow rates in the tempered water (annulus) system may result in the tempered water temperature exceeding 50 °C for short periods of time after an SCIC tempered water high temperature trip. This does not violate the safety function of this parameter limit provided the heater is de-energized when temperature exceeds 50 °C. This system is mode independent and is active throughout the MCO processes.

Process Upset and Accident Conditions  Both loss of electrical power and a seismic event are common mode failures that must be precluded from affecting the safety function of the SCIC. To meet this requirement, the design of the SCIC system ensures activation of an MCO ISO & PURGE or trip of the tempered water heater under process upset conditions (see Figure B2-27). Included in this protection are loss of electrical power and a seismic event that activate the relays for MCO ISO & PURGE and shutdown of the tempered water (annulus) system heater.

An excessive bay temperature also has the potential for adversely affecting the SCIC safety function. To preclude this situation, a high bay temperature trip is provided to initiate an MCO ISO & PURGE when the room temperature exceeds the parameter limit of 95 °F. The principle component that is affected by temperature is the MCO vacuum pressure transmitter calibration. Other components meet or exceed the maximum room temperature criterion of 115 °F.

Control Room Alarms  The SCIC system contains safety-class redundant annunciators that provide alarms for BAY ISO & PURGE, ANNULUS LOW LEVEL and PWC LOW FLOW. Activation of the ANNULUS LOW LEVEL or the PWC LOW FLOW alarms alerts the operator that corrective action is required. Although included on the safety-class annunciator, the PWC LOW FLOW alarm is not required for safe operation of the facility.

Seismic Safe Shutdown  The seismic trip is designed to detect a seismic event, isolate the MCO and actuate the SCHe system. In addition, a seismic event will shut off power to the tempered water (annulus) system heater. The parameter limit of 0.06 g was selected to ensure a seismic trip well in advance of a seismic event of significant magnitude to damage piping or equipment that is not part of the extended primary MCO boundary. The parameter level is at a point where most people would feel the effect but no damage would occur and is ~20% of the...
DBE level of 0.26 g. Two methods are used to preclude fault trips by the seismic monitoring equipment. First, the three sensors (each are triaxial) are located in three different process bays (2, 3, and 4), which precludes vibrations in one bay from tripping all three sensors. Second, each auctioneering logic circuit requires two of the three sensors to be in alarm to initiate a trip of each train. The seismic shutdown system is designed to performance category 3 criteria.

The accident analysis identified that a DBE could damage safety-class systems or components. Some components may not be operable after a seismic event to meet their safety function. To eliminate the need for seismic qualification, the seismic trip will perform the safety function (i.e., trip the MCO ISO & PURGE and de-energize the tempered water heater).

High Bay Temperature. High bay temperature sensor activation occurs when the mode selector switch is moved to the HEATUP position and continues throughout the remainder of the process. The process bay temperature monitoring system is designed to trip the SCIC system relay and activate an ISO & PURGE, if the bay temperature exceeds the parameter value. Design analysis identified the possibility that elevated temperatures in the process bay could result in instrumentation and control inaccuracy and malfunctions that could affect the function of safety-class systems. The bay temperature trip serves to mitigate temperature impacts on the two vacuum pressure transmitters PT1*08 and PT1*10. These devices have a calibration limit of 105 °F. In addition, to support assumptions in the fire hazard analysis related to combustible loading limitations, a maximum bay temperature of 95 °F was utilized to calculate safety-class instrumentation temperature protection from localized fires. Thus, an SCIC trip is initiated if bay temperatures exceed a 95 °F parameter limit. This parameter limit also protects the upper accident environmental temperature limit of 115 °F.

Nonqualified Signal Protection. The SCIC system provides non-safety class signals to the facility MCS for normal control, indication, and alarms. The signals from the SCIC system include all analog signals (flow and pressure), SCIC internal logic status, and trip and alarm contact status. All signals from the SCIC system are electrically isolated from any nonsafety system, including the MCS.

Sensor Error. Analysis has been performed for all sensors associated with functional operation of the SCIC system trip setpoints to allow for instrument error (see SNF-3091). Trip setpoints shown in Table B4-6 ensure critical parameters will not be exceeded for all functional requirements. For all analog signals coming from pressure and flow, an average of the signal is performed to remove instrument spikes that may occur.

Testing Functions. The local operating panels in each process bay provide test switches to perform operational tests on the following SCIC functions: tempered water (annulus) system high temperature trip, bay high temperature trip, and tempered water (annulus) low-level alarm. The test is enabled by a key switch on the panel. Once enabled, a trip can be simulated by pressing the appropriate test button. The bay high temperature test will isolate the MCO and start an SCHe system purge, the tempered water high temperature trip test will de-energize the tempered water heater, and the tempered water (annulus) low-level alarm will active the
safety-class annunciation in the control room. Testing associated with MCO isolation and SCHe system purge is normally conducted by using a jumper, located on the process hood support stand to provide a pathway between the process port connectors.

Other functional tests or calibrations require the use of a SCIC calibration and test computer unit that tests logic functions and allows for detailed calibration functions. The calibration and test computer is part of the SCIC system but is not normally connected to the SCIC PLC.

**Single-Failure Evaluation** The SCIC system has been designed to be single-failure proof and provides redundancy for all sensors and control modules. Single-failure analysis for the SCIC system logic controller was performed as a procurement requirement by the vendor. Single-failure analysis of the SCIC system has been performed and is documented in SNF-4290. Responses to an event are unchanged by failure of any one component due to the independent action of the second SCIC system instrumentation.

**Seismic Qualification** The whole of the SCIC system plus the sensors that provide input to the SCIC and output devices from the SCIC (valves and tempered water contactors), is seismically qualified with the following exceptions: The control room annunciator and mode control panels are not required to be qualified because these signals are protected from a seismic event by the seismic shutdown circuits. The design of the SCIC system eliminates any need for operator response to protect the MCO content against a seismic event. The loss of the annunciator after a seismic event strong enough to fail items in the control room will be obvious to the operator or to the site response team.

The SCIC system seismic trip setpoint and parameter limit are shown in Table B4-6. Process sensors (pressure and flow) are not seismically qualified for operability (only pressure boundary as required) because the seismic monitoring trip will satisfy the safety function after a seismic event. Seismically qualified local indication is provided for accident recovery monitoring. These components are not part of the SCIC system.

Engineering calculations demonstrating the adequacy of the performance category 3 SCIC system are listed in SNF-3001, *Cold Vacuum Drying Facility Supporting Data and Calculation Database*. In addition, the seismic trip portion of the SCIC system will place the MCO process system in a safe-shutdown mode if a major seismic event occurs. All active SCIC equipment that is required to maintain functional operability during and after a seismic event has been seismically qualified by shake-table testing. The safety-class cabinets are bolted to the process bay floor to meet performance category 3 seismic requirements.

**B4 3 1 5 Controls (Technical Safety Requirements)** The following assumptions associated with the SCIC system require TSRs to ensure performance of its safety functions:

- The SCIC system is operational before the MCO process port connections are made.
The other systems that provide signal input to the SCIC system (i.e., pressure transmitters, flow-indicating transmitters, tempered water high temperature switches, cask-MCO annulus water low-level switches, process bay temperature switches, and seismic trips) are operational.

- The seismic trip component is operational when an MCO is being processed.
- The process bay temperature monitor in a process bay is operational when an MCO is being processed in that bay.
- Standard calibrations of the SCIC system and functional testing of each SCIC train for ISO & PURGE trip shall be performed at periodic intervals (see setpoints in Table B4-6).

### B4.3.2 Safety-Class Helium System

**B4.3.2.1 Safety Function** The overall safety function of the SCHe system is to preclude accumulation of flammable concentrations of hydrogen and oxygen within the MCO, and to provide heat transfer within the MCO and a pressure vent pathway to the local exhaust system. The accident analyses presented in Chapter B3.0 identify four specific event scenarios in which the SCHe system is credited: thermal runaway reaction, MCO overpressurization, internal hydrogen explosion, and external hydrogen explosion.

The SCHe system performs the following safety functions for these accidents:

- **MCO overpressurization** — Upon actuation by the SCIC system or by manual operator action, the SCHe system provides a pressure-regulated discharge flow path from the MCO to the local exhaust system. In addition, the SCHe supply system pressure control valve and rupture disk in each SCHe system train protects the MCO from helium overpressurization by eliminating the possibility that the helium bottle pressure could be applied directly to the SCHe system lines.

- **Thermal runaway reaction** — When actuated by the SCIC system or by manual operator action, the SCHe system repressurizes the MCO to provide enhanced cooling of the fuel and thereby reduce the reaction heating effects during a thermal runaway reaction accident. The SCHe system also provides a pressure vent path to the local exhaust system for the helium purge of hydrogen produced in the MCO to ensure heat transfer capability within the MCO. The SCHe system, in conjunction with operation of the safety-class tempered water (annulus) system, are key systems for placing the MCO in a stable condition following a process upset.
- Internal hydrogen explosion — When actuated by the SCIC system or by manual operator action, the SCHe system pressurizes and purges the MCO to prevent the formation of flammable or explosive concentrations of hydrogen and oxygen inside the MCO. Pressurizing the MCO also prevents air ingress.

- External hydrogen explosion — When actuated by the SCIC system or by manual operator action, the SCHe system provides a pressure-regulated discharge flow path from the MCO to the local exhaust system, with backflow prevention to preclude flammable mixtures within the SCHe vent lines. The discharge flow path from the MCO to the local exhaust system is delayed more than one minute from SCHe actuation to allow standby power startup of the local exhaust fans if normal power is not available.

The SCHe system is designated a safety-class system because the unmitigated consequences of the thermal runaway reaction and the MCO overpressurization accidents could exceed offsite release limits. The hydrogen explosion accident scenarios that the SCHe system is credited with mitigating or preventing have consequences that exceed onsite risk evaluation guidelines but do not exceed offsite release limits. The SCHe system is credited with a safety-significant function for those accidents.

**B4 3 2 2 System Description** The SCHe system is a dedicated safety system that has no direct processing functions. The SCHe system is controlled and powered by the SCIC system. The SCHe system is automatically actuated by (1) loss of electrical or pneumatic power to its fail-open isolation valves or (2) the SCIC system in response to predetermined process setpoint deviations, excessive process bay temperature, or a seismic event. There are independent and identical SCHe systems for each processing bay at the CVDF. The piping and component configurations for the SCHe system are described in the composite CVDF engineering flow diagram shown in Figure B2-17 and in SNF-3068 *SNF Project Cold Vacuum Drying Facility Safety Class Helium System Design Description*.

The SCHe system for each process bay consists of four separate SCHe trains (purge A, purge B, purge C, and purge D). Each train consists of helium supplies, a rupture disk, pressure control valves, and fail-open electropneumatically operated isolation valves with associated piping and instrumentation. Purges A and B are configured identically and Purges C and D are configured identically. Both system pairs operate in parallel. Purges A and B are connected to the MCO inlet (long process tube) valve through the process hood and flexible tubing. Purges C and D connect to the filtered process exit port (port 2, which is HEPA-filtered) valve but are configured differently from purges A and B by being connected to the process vent system through pressure control valves. The two purge systems are mounted on individual panels and located back-to-back (e.g., purge A and purge C are mounted on individual panels and located back-to-back). The other two purge systems (purge B and purge D) are mounted in a similar manner. Panels and helium bottles for parallel systems are physically separated by mounting them to the east and west of the process equipment skid (PES), as shown in Figure B2-21. The SCHe bottles are mounted at each panel, and the SCHe purge systems have cage protection. The panels
that contain the valves, piping, and instrumentation are mounted above the helium bottles and are welded to the panel support structure.

The SCHe system uses two independent helium sources. Each SCHe train (A, B, C, and D) receives helium from the general-service helium supply (tube trailers) and from its dedicated SCHe bottles. The SCHe bottles are 15 ft³ compressed gas cylinders with a full capacity of 244 standard ft³ of helium at 2,480 lb/in² gauge. Pressure settings for the SCHe and general-service helium lines that connect to the MCO long axial process tube connectors are different than the pressure settings for the SCHe and general-service helium lines that are connected to the MCO filtered process exit port connectors. For each train (A, B, C, and D), the pressure regulator for the general-service helium is set higher than the regulator for the SCHe system bottles. This pressure differential allows helium to preferentially flow from the normal general-service helium process supply as long as its pressure remains above the pressure regulator setpoint for the SCHe bottles. When the general-service helium supply becomes exhausted or interrupted, helium is then provided to the system from the SCHe bottles. A minimum 30 minutes of purge volume is provided by the SCHe system bottles to purge the MCO. After this time, the MCO remains pressurized to preclude air leakages.

The general-service helium pressure in purge trains A and B is set at 15 lb/in² gauge (14 lb/in² gauge when supplied by the SCHe bottles), while the general-service helium pressure in purges C and D is set at 3 lb/in² gauge (2 lb/in² gauge when supplied by the SCHe bottles). Purges C and D are also connected to the process vent system through two pressure control valves in series set at 10 lb/in² gauge. These pressure differentials allow the general-service helium to preferentially flow from the 15 lb/in² gauge purges A and B through the MCO inlet, down the long axial process tube, upward past the fuel baskets, out through the HEPA-filtered exit, into purges C and D, and out into the process vent system through the 10 lb/in² gauge pressure control valves. Should purges A and B become blocked, broken, or empty such that the system pressure falls to the regulator setpoints in purges C and D, helium would be supplied to the MCO by the lower pressure purges C and D. In this case, helium would not vent through the process vent system because the system pressure is lower than the 10 lb/in² gauge relief valve setting. Helium would either vent through the break in purge A or B, diluting the hydrogen concentration in the MCO, or the MCO would stay pressurized above atmospheric pressure, preventing oxygen ingress. It is important to note that the pressure settings provide a delay of at least 1 minute to allow sufficient time for the backup power to activate the local exhaust fans in the event of loss of power (SNF-3001, Calculation CVD-16 [Rev. 3]). This provision ensures that hydrogen generated in the MCO will be adequately diluted in the local exhaust.

There are eight major interfaces between the SCHe system and the CVDF, as described in Table B4-7.

The SCHe system is pressurized and available any time the MCO is connected to the VPS process connectors. It is actuated by loss of electrical or pneumatic power to the fail-open isolation valves in each individual purge system or by signal from the SCIC system (see Section B4 3 1 for specific conditions that cause the SCIC system to actuate the SCHe system).
## Table B4-7 Safety-Class Helium System Interfaces

<table>
<thead>
<tr>
<th>Interfacing system or facility</th>
<th>Interface function</th>
</tr>
</thead>
<tbody>
<tr>
<td>Process hood support stand and PWC/VPS-MCO connectors</td>
<td>Interconnect the SCHe system to the MCO through safety-class flexible hoses</td>
</tr>
<tr>
<td>CVDF general-service helium supply</td>
<td>Connects the SCHe to a non-SCHe supply</td>
</tr>
<tr>
<td>Safety-class instrumentation and control system</td>
<td>Initiate actuation of the SCHe system and isolation of the MCO from the VPS and the PWC systems, also provides an alarm to the control room on SCHe actuation</td>
</tr>
<tr>
<td>CVDF electrical system</td>
<td>Provides 120 V AC, single-phase signal power to the SCHe system isolation valves via the SCIC system relays</td>
</tr>
<tr>
<td>CVDF instrument air system</td>
<td>Provides pressurized air to close the SCHe system gas-operated valves (fail open on loss of air or electric power)</td>
</tr>
<tr>
<td>CVDF local exhaust system</td>
<td>Provides dilution of the SCHe system vent stream during helium purge</td>
</tr>
<tr>
<td>CVDF building</td>
<td>Provides mounting structure (floor) for safety-class SCHe panels and helium bottles</td>
</tr>
<tr>
<td>Monitoring and control system</td>
<td>Provides non-safety class monitoring of the SCHe system</td>
</tr>
</tbody>
</table>

CVDF = Cold Vacuum Drying Facility  
MCO = multi-canister overpack  
PWC = process water conditioning system  
SCHe = safety-class helium  
SCIC = safety-class instrumentation and control  
VPS = vacuum purge system

Actuation of the SCHe system by the SCIC system also coincides with automatic closure of the safety-class isolation valves for the VPS, the PWC system, and general-service helium system on the process piping leading to or from the MCO as well as opening of the SCHe system isolation valves. Once initiated, the system remains actuated until reset by an operator at the SCIC panel.

The SCHe system contains instrumentation and alarms to monitor critical system parameters. Local safety-class pressure gauges are provided at each SCHe supply bottle for checking helium bottle pressure. General-service alarms for low bottle pressure (1,700 lb/in² gauge set point) are sent to the MCS. The 1,540 lb/in² gauge helium bottle pressure will achieve 30 minutes of SCHe flow to meet the system requirements. When an SCIC ISO & PURGE trip occurs, a safety-class alarm is activated in the control room and the SCHe system is actuated. Other instruments for local and remote monitoring (flow valves and flow switches) are nonsafety-class but are designed to require multiple failures to avoid impacting the safety-class systems.
B4 3 2 3 Functional Requirements  The functional requirements needed for the SCHe system to perform its safety functions are as follows

- **Helium Injection** — Provide helium into the MCO in order to prevent flammable concentrations of hydrogen and oxygen from forming inside the MCO. Maintain a positive pressure to prevent air ingress to the MCO (assuming no line break) by injecting helium.

- **Vent Path** — For an isolated MCO that is being purged with helium because of SCHe activation, provide a pressure vent path from the HEPA-filtered side of the MCO to the local exhaust system.

- **Initial Purge Delay** — The supply and discharge pressure control valve settings shall accomplish an appropriate pressure differential that results in at least a 1 minute delay between SCHe actuation and vent path discharge to the process vent system. This allows the process bay local exhaust heating, ventilation and air conditioning (HVAC) and process vent system to restart on diesel standby power to reestablish dilution flow before SCHe discharges into the ventilation system ductwork.

- **Pressure Monitoring** — Contains sufficient local pressure monitoring capability for each helium bottle to ensure verification of the required operation.

- **System Isolation** — To be capable of isolation from the VPS general-service helium, and PWC system while functioning, in order to ensure venting into the local exhaust system.

- **Helium Flow** — To be capable of supplying sufficient helium flow to preclude explosive hydrogen concentrations in the MCO and in the vent path to the local exhaust system. A minimum supply of helium at a minimum flow rate of 14 standard ft³/min is required to flow for 27 minutes.

- **Purge Duration** — Purge out any air that may have entered the MCO, thereby precluding accumulation of flammable concentrations of hydrogen and oxygen.

- **Pressure Control and Relief** — Provide pressure control and relief in the helium supply line to avoid a situation where the helium bottle pressure can be applied directly to the MCO and exceed the 150 lb/in² gauge design pressure of the SCHe lines and the MCO if the bottle pressure regulator fails.

- **Accident and Environmental Conditions** — To be capable of operation under accident conditions (including loss of power) and worst-case surrounding environmental conditions.
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- **Helium Purity** — Provide helium of sufficient purity to be acceptable for mitigation of the thermal runaway accident and prevent internal MCO flammable conditions

- **System Operability, Calibration, and Testing** — To be operable before bulk water is drained from the MCO, ascertain that the safety-class system is functioning as required and that setpoints for all safety-class pressure relief valves are as specified

- **Single-Failure Evaluation** — To be single-failure proof

- **Seismic Qualification** — Provide a leak-tight SCHe system by use of seismically qualified valves, piping, hoses and other components, and be sufficiently secured to withstand a seismic event and remain operable following the DBE

**B4 3 2 4 System Evaluation**  The SCHe system is a safety-class design that is activated by the SCIC system in response to process upset conditions, loss of power, or operator action. The SCHe system is designed to operate during all of the postulated events it is intended to prevent or mitigate, including the DBE event.

**Helium Injection**  The SCHe system uses fail-open, electropneumatically operated isolation valves that (1) provide reliable, leak-tight operation, (2) fail open upon signal from the SCIC system by de-energizing the SCIC output relay that controls power to the valves, loss of instrument air or loss of electric power, and (3) remain open until the SCIC is manually reset. Upon activation, the SCHe system provides helium to the MCO via four lines (two lines into each MCO open port). Initially, the SCHe system will use helium from the pressurized general-service helium system, but if that system is unavailable, helium will be supplied from the pressurized bottles in the SCHe system. The capacity for the SCHe system helium bottles to provide MCO purge is based on the worst-case 1-in safety-class line break inside the isolation valve boundary. Engineering calculations that support the analyses of the worst-case scenarios and that form the basis for appropriately sizing the SCHe system SSCs are listed in SNF-3001, CVD-16. If helium is not available from the general-service helium system, the operator has about ½ hour to respond to a major line break. Even if the MCO eventually goes to atmospheric pressure, diffusion of air into the MCO is time dependent, which allows considerably more time for operator response. If there is no major line break or system leaks, one set of SCHe bottles will provide purge through the MCO and the second set of bottles will maintain pressure on the MCO for at least 96 hours, which will allow facility recovery actions. The positive pressure is sufficient to prevent ingress of air into the MCO.

**Vent Path**  There are dual SCHe system lines connected to the VPS line from the MCO filtered process exit port that provide redundant vent paths through the HEPA-filtered MCO port connection to the local exhaust system. Settings on the pressure control valves ensure a delay of at least 1 minute for venting to allow time for the standby power to activate the local exhaust fans. If there is a power failure, this time delay has been shown to be sufficient to prevent hydrogen within the MCO from reaching a flammable condition in the local exhaust system.

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Engineering calculations that demonstrate the required pressure settings for the SCHe and general-service helium systems pressure control valves accomplish the time delay are listed in SNF-3001 Calculation CVD-33.

Initial Purge Delay The general-service helium supply pressure to SCHe purge trans A and B is set at 15 lb/in² gauge (14 lb/in² gauge when supplied by the SCHe bottles) SCHe trains C and D are each connected to the process vent system through two pressure control valves in series set at 10 lb/in² gauge. These pressure differentials allow the helium to flow from purge trains A and B through the MCO inlet, down the long axial process tube, upward past the fuel baskets, out through the HEPA-filtered exit, into purge trains C and D, and out into the process vent system through the 10 lb/in² gauge pressure control valves. The pressure settings provide a delay of at least 1 minute before discharge to the process vent system to allow sufficient time for the backup power to activate the local exhaust fans in the event of loss of power (SNF-3001, Calculation CVD-16 [Rev 3]). This provision ensures that hydrogen generated in the MCO will be adequately diluted in the local exhaust, preventing a potential external hydrogen explosion initiated by a loss of normal electrical power.

Pressure Monitoring The line from each SCHe system helium bottle contains safety-class pressure indicators. A minimum helium bottle pressure parameter of 1,540 lb/in² gauge is required to provide sufficient volume for 30 minutes of helium purge. Engineering calculations demonstrating that this pressure provides adequate purge flow are provided in SNF-4301 (see Purge Duration below).

System Isolation The SCHe system is designed to be actuated by the SCIC system and, once actuated, to remain independent of all CVDF support utilities for functional operation. Proper operation of the SCHe system requires closure of the safety-class isolation valves for the VPS, general-service helium system, and PWC system. Following an SCIC trip condition requiring SCHe actuation, the isolation valves in the other systems switch to the closed position (until reset) to establish and maintain the pressure boundary required for the SCHe system safety function.

There are four SCHe system isolation valves, two connect via VPS lines to the long process tube connector and two connect via VPS lines to the MCO filtered exit port connector. These valves are 1-in stainless steel, gas-operated valves rated at 150 lb/in² gauge and are a ball-type design that use a cylinder and piston air actuator to rotate the ball shaft. Each valve also has local position indication. Each safety-class isolation valve is controlled by an associated safety-class electronic solenoid valve. Normally, the solenoid valve is energized in the open position to allow instrument air supplied to the SCHe system isolation valve actuator, keeping the SCHe isolation valves closed. Under SCIC trip conditions, the SCIC de-energizes the solenoid, which in turn closes the solenoid valve, blocks the instrument air flow, and allows existing air in the inner chamber to vent out, which then causes the isolation valve to switch to its fail-safe open state via a spring actuator mechanism. The solenoid valve operation is fail-safe in the respect that a loss of electrical power or loss of instrument air would cause the desired safety function to occur (i.e., the SCHe isolation valves would open). Safety-class filters are required on...
the instrument air lines to the actuators on the isolation valves to prevent buildup of foreign material and preclude a potential isolation valve malfunction. Moisture in the instrument air line is eliminated to improve overall reliability of value actuators by use of a general-service dryer in the instrument air system.

**Helium Flow** The SCH e system is designed with a pressurized source of helium piping, control valves, and instrumentation and controls that deliver helium to the MCO and provide a pressure vent path for MCO gases. The SCH e supply is designed to provide more than 1.4 standard ft³/min of helium flow that is sufficient to preclude air ingress and prevent accumulation of flammable concentrations of oxygen and hydrogen in the MCO (i.e., purge air out of the MCO so that it is pressurized and does not have sufficient oxygen to reach flammable conditions).

**Purge Duration** A minimum duration of helium flow is required from the SCH e bottles to ensure that oxygen is adequately purged out of the MCO to preclude flammable conditions. Purge duration calculations provided in SNF-4301 demonstrate that purge flow at the minimum SCH e flow rate of 1.4 standard ft³/min accomplishes this safety function in 27 minutes. Bottle pressure calculations at the maximum supply flow rate (design feature [SNF-3001, Calculation CVD-161]) of 4 standard ft³/min and 30-minute duration require a minimum bottle pressure of 1,540 lb/in² gauge.

**Pressure Control and Relief** Helium is supplied from each SCH e bottle through a pressure control valve set at 25 lb/in² gauge. A 125 lb/in² gauge rupture disk that vents to the room is located between the tank pressure control valve and the system pressure control valve. This configuration prevents overpressurization of the MCO by the SCH e system. The general-service helium system also has redundant safety-class relief valves set at 25 lb/in² gauge, well below the MCO design pressure to prevent overpressurization from this source of supply.

**Accident and Environmental Conditions** The independence of the SCH e system design assures that the system will be available under conditions where accidents may occur. The SCIC system provides control and power to the SCH e system. The only power supplied to the SCH e system is to activate the isolation valves that are normally held in the closed position (i.e., they fail in the open position). Response to accident situations is assured by the interrelationship with the SCIC system that monitors processing conditions and activates the SCH e system if process upsets occur. Also, the SCH e system is designed to environmental condition B of the safety equipment list and is operable to 115 °F (at 22% humidity) well above the parameter limit of 95 °F for the SCIC system bay high temperature sensor. The SCIC system will initiate an ISO & PURGE if the bay temperature setpoint is exceeded thereby isolating the MCO and activating the SCH e system.

**Helium Purity** Purity of the SCH e system bottles, >99% (as derived from SNF-2770), is controlled through the vendor quality assurance program, tank receipt inspection under the CVDF quality assurance program, and color markings on the helium bottles.
System Operability, Calibration, and Testing  The SCHe system is designed to be functionally tested for operability. Draining of bulk water is the initial step in the cold vacuum drying operation; therefore, the SCHe system must be operable prior to initiation of the process. In addition, the SCHe system has redundancy, is fail-safe, and is routinely surveilled and maintained. SCHe components are designed to be functionally tested for operability given a simulated process condition input. System availability is verified by surveillance of the system’s supply pressure before enabling the system for each MCO process cycle. Periodic testing of SCHe safety-class components is dictated by the requirements of the individual components according to the respective manufacturer’s recommended schedules and practice, and is administered by controlled procedures for all safety-class and safety-significant SSCs.

Recalibration of signals from the pressure transmitters to the SCIC system is maintained according to the respective manufacturer’s recommended schedule and practice. The SCHe system is designed with calibration and test connections to enable periodic measurement and calibration of all setpoints and adjustments that affect the manner in which the SCHe system performs.

Single-Failure Evaluation  For single-failure considerations, the SCHe system has been designed with parallel and redundant systems and to operate without external power in a fail-safe manner. The SCHe system is actuated by the SCIC system, which is a safety-class system itself. The SCIC system uses redundant and independent safety-class sensors on process systems and monitors process bay temperatures and seismic trip sensors (which include auctioneering, 2-out-of-3 logic to preclude fault trips), to actuate the SCHe system.

The SCHe system safety-class system has been evaluated by single-failure analysis (SNF-4290). All SCHe system components that perform the helium injection and venting functions are designated safety-class and meet performance category 3 seismic, redundancy, and physical separation requirements. This includes redundant helium supply paths to each MCO port and redundant venting paths that allow the system to function following a single-point blockage or failure. Single random failures of the SCHe system have been shown to be tolerable. The MCS also provides surveillance of the general-service helium system components, thereby providing added assurance that any system malfunction will be detected and responded to by operators.

Seismic Qualification  The design provides physical separation of redundant purge trains of the SCHe system. SCHe system pairs are mounted to the facility floor via support stands designed to performance category 3 criteria, and therefore remain structurally sound following design basis natural phenomena events. Also, the process bay structure is designed and constructed to performance category 3 for natural phenomena hazards.

The SCIC system has sensors for detecting a seismic event and for indicating an excessive bay temperature that will activate the SCHe system if conditions are not acceptable for operation. All supporting systems that perform sensing, logic, actuation, and isolation functions necessary for the SCHe system to perform its functions are also designated safety-class — meeting performance.
category 3 seismic redundancy, and physical separation requirements. All components are mounted to structural components meeting performance category 3 requirements. Seismic calculations for the SCHe system components and piping are listed in SNF-3001.

**B4 3 2 5 Controls (Technical Safety Requirements)** The following assumptions associated with the SCHe system require TSRs to ensure performance of their safety functions:

- The SCHe system including vent path is operable prior to making process connections to the MCO.
- Functional tests and leak tests of the VPS, general-service helium system, deionized water system, and PWC system isolation valves have been performed on an appropriate schedule consistent with the respective manufacturer’s recommended schedule and practice.
- The purge train A and B supply pressure control valve settings in conjunction with the purge train C and D pressure control valve settings to discharge to the process vent system will accomplish at least a 1 minute delay prior to discharge of helium purge flow to the process bay local exhaust HVAC and process vent system.
- The helium gas bottles are of a purity of >99%.
- The fail-open isolation valves for the SCHe system will function on demand.
- The SCHe bottles contain a sufficient quantity of helium before each drying cycle begins (minimum pressure of 1700 lb/in² gauge including error and readability).

**B4 3 3 Cask-Multi-Canister Overpack Safety-Class Components**

In addition to their normal functions, the cask-MCO components perform safety-class functions for prevention of the MCO overpressurization accidents, thermal runaway accident, and criticality. Safety-significant functions are performed for prevention of the gaseous release, external hydrogen explosion, and internal hydrogen explosion accidents. These components are associated with the hydraulic boundary of the tempered water (annulus) system interface and the pressure boundary for the MCO. The cask-MCO also provides a safety-significant function of shielding during normal operations for worker safety. Section B4 4 1 4 provides specific positioning requirements for the cask-MCO trailer to protect assumptions in the seismic analysis.

**B4 3 3 1 Safety Function**

**Transportation Cask** Following receipt at the CVDF, the cask headspace is vented, the cask lid is removed, and a CVDF process hood-seal ring assembly is bolted to the cask. The seal ring isolates the cask annulus space via two sealing bladders provides the tempered water.
(annulus) system outflow connection, and provides a connection point for instrument air supply for annulus drying. The seal ring is described in Section B4.3.4 as a component of the tempered water (annulus) system (Figure B4-1). The cask safety function for the thermal runaway reaction accident is to contain water in the cask annulus for prevention and mitigation of the accident. The tempered water supply is connected to the quick-disconnect drain fitting, which is safety-class to ensure function and integrity of the connection.

**Multi-Canister Overpack** The MCO is designed to contain and confine its contents during all normal operations and accident conditions and to maintain fuel elements and fuel fragments in a critically safe geometry throughout the MCO's design life both before and after being subjected to the DBAs. The MCO also provides the inner hydraulic boundary of the cask-MCO annulus and one of the two sealing surfaces for the cask-MCO seal ring. The MCO is credited with prevention or mitigation functions for the gaseous release, external hydrogen explosion, internal hydrogen explosion, thermal runaway reaction, and MCO overpressurization accidents. The MCO performs the following safety functions for those accidents:

- **MCO overpressurization** — A passive function of the MCO is to maintain its pressure boundary at the design rating of 150 lb/in² gauge and to maintain the structural integrity of the cask-MCO annulus required for tempered water to provide adequate heat transfer from the MCO.

- **Thermal runaway reaction** — A passive function of the MCO is to maintain its confinement boundary and the structural integrity of the cask-MCO annulus required for adequate heat transfer from the MCO.

- **External hydrogen explosion** — The function of the MCO is to prevent hydrogen from escaping in an uncontrolled fashion and creating explosive conditions external to the MCO by maintaining its structural integrity and leak tightness.

- **Internal hydrogen explosion** — The function of the MCO is to prevent air ingress that could result in explosive conditions by maintaining its structural integrity and leak tightness and to maintain the structural integrity of the cask-MCO annulus required for adequate heat transfer from the MCO.

- **Gaseous release accident** — The MCO is credited with confinement of radioactive materials within its pressure boundary.

The prevention of a criticality accident is also a safety-class function, therefore the MCO and Mark IA fuel baskets function as safety-class components provide geometry control to prevent a criticality (see Chapter B6.0).

In addition to providing containment, the MCO shield plug is also designed to minimize radiation doses (safety significant) to personnel during processing and handling operations.
B4 3 3.2 System Description

Transportation Cask The MCO transportation cask is a vertical, cylindrical, stainless steel vessel. The cask has an outside diameter of 39.81 in and a height of 170.25 in. The overall packaging assembly, including the lifting device, has an outside diameter of 43.83 in and an overall height of 190.25 in. The cask cavity diameter is 25.19 in and is 160.50 in deep. The cask consists of a forged 7.31-in-thick 170.25-in-long stainless steel cylinder with an integrally welded stainless steel bottom head that is 6.13 in thick. A quick-disconnect fitting is positioned near the bottom of the cask and communicates with the interior cavity. This safety-class fitting provides the connection with the tempered water (annulus) system and is protected with a port cover when not in use. Refer to Figure B2-3 for the design of the MCO and cask package.

The cask lid is forged stainless steel with a 3.5-in-thick top and 3-in sides, an outer diameter of 35.5 in, an inner diameter of 25.5 in, and a height of 11 in. The lid has a 4-in-tall, 4.16-in-wide flange at its base, which makes the diameter of the lid 39.81 in at the flange. The base of the lid has an O-ring groove that has an inner diameter of 31.57 in. The interior of the base of the lid also has a 2.16-in-wide 1.09-in-tall notch that mates with a similar-sized extension in the cask shell. The lid is bolted to the cask body with 12, 1.5-in-diameter bolts that are arranged on a circle with a 36.44-in-diameter. A single, butyl, O-ring seal forms the containment boundary between the cask body and lid. Lid installation is guided by two alignment pins that are integral to the cask body.

Multi-Canister Overpack The MCO provides a confinement system for cold vacuum drying processing and storage of N Reactor SNF currently located in the K Basins. The MCO is constructed from austenitic stainless steel with high resistance to corrosion from all aspects of the environment to which the system is expected to be exposed during its lifetime. From loading at the K Basins to placement in interim storage, the interior of the MCO sees three distinct thermal-hydraulic conditions: flooded with water, vacuum or near-vacuum conditions, and dried and backfilled with helium gas. While a combination of these three distinct conditions will exist as an MCO passes from one process step to the next, the safety basis of the MCO is principally established using steady-state and transient analysis under each of these distinct operating conditions (HNF-SD-SNF-SARR-005). While at the CVDF, the MCO will see all three distinct operating conditions.

The major MCO components are the MCO shell, bottom assembly, canister collar, shield plug with attached long process tube, pressure rupture disk, locking and lifting ring, lifting cover cap, fuel baskets, and scrap baskets.

The overall dimensions of the MCO are 24 in in diameter by 160 in in height. The wall thickness is 0.5 in. The shell is welded to a stainless steel bottom forging 24 in in diameter. A 16-in-thick shield plug (including filter chamber and filter guard plate) is inserted into the top of the MCO. The MCO top shield plug is held into the MCO shell by a threaded assembly called the locking and lifting ring. For reference see Figure B2-15 for the design of the MCO.
The cylindrical shell, bottom assembly, canister collar, shield plug, and locking and lifting ring form the primary pressure-retaining boundary for the SNF.

The MCO shell provides confinement of the fuel during handling and storage. MCO materials, fabrication, welding, and examination are in accordance with the requirements of the ASME Boiler and Pressure Vessel Code (ASME 1995) Section III (HNF-SD-SNF-SARR-005). The MCO also provides support for the shield plug and fuel baskets for stacking, lifting, and handling. The top and bottom baskets may also be scrap baskets, with a design that includes a series of reed fins around it that contact the MCO inner wall to reduce gas flow up the annulus and preferentially direct it through the basket. The scrap baskets are designed to enhance thermal conductivity to the MCO. The bottom assembly welds are made during fabrication of the MCO.

The shield plug is a multifunctional component of the MCO. It provides a mechanical confinement boundary until the MCO cover cap is welded in place. The shield plug also provides for axial radiation shielding to allow personnel access to the top of the MCO for closing the MCO and for performing the drying and processing functions. The shield plug retains the main seal, which seals between the MCO collar and the shield plug. The shield plug has four ports that connect to four penetrations, two of which are connected to an internal HEPA filter. One of the two HEPA-protected penetrations allows for the filtered release of gases from the MCO during vacuum drying. The other HEPA port is blanked. Another penetration connects to the long process tube, which reaches to the bottom of the MCO, aiding in the removal of water during draining. The final penetration is for the safety-class rupture disk with a nominal 150 lb/in² gauge rating. The design of the shield plug subassembly is shown in Figure B2-16.

When the MCO is positioned inside the cask, a 0.595-in.-wide annulus is formed between the cask and the MCO. During fuel loading at the K Basins, this annulus is filled with water, which completes the heat transmission path for the nuclear decay heat and chemical corrosion heat generated by the fuel. At the CVDF, tempered water from the tempered water (annulus) system is circulated through the annulus to adjust the temperature of the fuel as one of the process parameters.

B4.3.3.3 Functional Requirements

**Transportation Cask** To perform its safety function of containing water in the cask-MCO annulus during the internal and external hydrogen explosion, thermal runaway, and MCO overpressurization accidents, the cask must remain structurally stable and not leak. The cask drain fitting must retain its integrity as a leaktight connection to the tempered water supply piping. The cask must be precisely located in the process bay such that seismic-induced motion of the cask and transporter do not result in damage to the process hood support stand piping or connections to the MCO.

**Multi-Canister Overpack** To perform its safety functions of containing water in the cask-MCO annulus and preventing water leaks to the fuel inside, the MCO is also required to remain structurally stable and not leak. It must also retain its integrity as a leaktight boundary for...
gaseous material. The 150 lb/in² gauge (nominal) rupture disk function is credited in the accident analysis as a design feature of the MCO.

**B4 3 3 4 System Evaluation**

**Transportation Cask** The cask because of its massive construction and design as safety class for transportation (HNF-SD-TP-SARP-017) will be able to maintain its dimensions and not collapse the annulus. Likewise, a leak from the cask body is improbable. The lower cask port fitting is tested and installed as safety-class to ensure its integrity.

**Multi-Canister Overpack** The anticipated normal MCO operations at the CVDF include being closed and static (no connections, cask closed), connecting and venting, water draining, purging with helium, and vacuum drying at temperatures below 50 °C (122 °F). The integrity of the MCO at the time of fabrication was verified by hydrostatic leak testing at the vendor. Verification that the fuel is sufficiently dry will be performed. Leak tests will be performed on the MCO connections and plug valves after they are closed. The 150 lb/in² rupture disk is safety-class. Engineering evaluation in HNF-SD-SNF-SARR-005, *Multi-Canister Overpack Topical Report* demonstrates the adequacy of the rupture disk to perform its design functions.

**B4 3 3 5 Controls (Technical Safety Requirements)**

**Transportation Cask—MCO** The following assumption associated with the cask—MCO requires TSRS to ensure performance of the safety function. It also is assumed that the safety-class lower cask port connection for tempered water supply has been installed in all casks as a retrofit activity before any fuel transport to CVDF.

- Design pressure of the MCO is 150 lb/in² gauge and the rupture disk is installed properly and is functional (not plugged by a blind cover plate).

**B4 3 4 Tempered Water (Annulus) System Safety-Class Components**

**B4 3 4 1 Safety Function** As described in Chapter B3 0, the thermal runaway reaction accident could occur if heat removal from the MCO is not sufficient to prevent fuel elements and scrap fuel from heating up as a result of decay heat and chemical reaction heat. As the temperature increases, the chemical reaction rate increases, producing more hydrogen and heat. To prevent such a scenario, heat must be dissipated from the MCO by the tempered water (annulus) system in conjunction with general-service helium and/or SCHe system operations. Water in the annulus between the MCO and cask is required to adequately transfer heat from the MCO to the tempered water (annulus) system. The overall safety functions of the safety-class components of the tempered water (annulus) system that prevent a thermal runaway reaction (in conjunction with the SCHe system) are to maintain the level and supply temperature of the water in the cask—MCO annulus within design limits.
The tempered water (annulus) system provides (1) safety-class functions for the thermal runaway reaction and the MCO overpressurization accidents, and (2) safety-significant functions for the internal hydrogen explosion accident. The tempered water (annulus) system performs the following safety functions for those accidents:

- **MCO overpressurization** — The safety-class functions of the tempered water (annulus) system are required to protect the analytical assumptions for heat dissipation from the MCO, by maintaining water in the annulus between the cask and the MCO. The system also provides a high inlet water temperature signal and a low annulus water level signal to the SCIC system. The SCIC system terminates power to the tempered water (annulus) system heater, which ceases heat input to the MCO. The SCIC system also provides a safety-class low annulus water level alarm to the operators for corrective action. Maintaining heat transfer out of the MCO will also reduce hydrogen generation in the MCO. The tempered water (annulus) system retains annulus water through piping integrity and antisiphon valves.

- **Thermal runaway reaction** — The functions of the tempered water (annulus) system to prevent the thermal runaway reaction accident are to provide for heat dissipation from the MCO by maintaining water in the annulus between the cask and the MCO. The system also provides a high inlet water temperature signal and a low annulus water level signal to the SCIC system. The SCIC system terminates power to the tempered water (annulus) system heater, which ceases heat input to the MCO. The tempered water (annulus) system provides a safety-class low annulus water level signal to the SCIC system, which alarms to alert the operators for corrective action. The tempered water (annulus) system retains annulus water through piping integrity and antisiphon valves, and the system design provides the capability for manual refill.

- **Internal hydrogen explosion** — The safety-significant functions of the tempered water (annulus) system to prevent the internal hydrogen explosion accident are to provide for heat dissipation from the MCO by maintaining water in the annulus between the cask and the MCO. The system also provides a high inlet water temperature signal to the SCIC system, which terminates power to the tempered water (annulus) system heater, and detects a low annulus water level, which alerts operators to manually refill the annulus. These actions cease heat input to the MCO and maintain the cask-MCO annulus water level, which reduces hydrogen generation within the MCO. Piping integrity and antisiphon valves in conjunction with the tempered water (annulus) system design ensure the annulus water is retained.

**B4 3 4 2 System Description** Two water lines are connected to the cask-MCO, one near the bottom of the cask, which serves as the tempered water inlet, and one in the cask-MCO seal ring, which serves as the annulus outlet. All tempered water (annulus) system components, from the piping between the cask ports and the valves and piping components on the process hood support stand (up to and including the antisiphon valves), form the safety-class primary barrier to prevent...
loss of annulus water  The cask inlet flex line is encased in an outer 3-in diameter flex line that forms a double-walled configuration to an elevation above the level of the fuel in the MCO  The lower end of the double wall pipe is connected to the cask via a special connector  All of these components including the quick disconnect for the supply line on the bottom of the cask, are classified as safety class  The remaining components of the tempered water (annulus) system (those on the pump side of the antisiphon valves and the connections of the flexible lines to the PES) are general-service components  Failure of the general-service equipment cannot lead to a loss of the cask-MCO annulus water  The general-service tempered water (annulus) system components include the inlet and return piping to the antisiphon valves, the system pump, heater, system cooler, holding tank and associated valves, indicators, and switches  Refer to Figure B2-17 and SNF-3085, SNF Project Cold Vacuum Drying Facility Tempered Water and Tempered Water Cooling System Design Description for more detailed information

The seal-ring assembly described in Section B4.3.3 forms part of the hydraulic boundary between the cask and the MCO  The cask-MCO seal ring is considered a part of the tempered water (annulus) system but is attached to the process hood  The seal ring's function is to form a water- and air-tight seal in two locations  (1) between the cask shell and the seal ring on a horizontal plane and (2) between the MCO and the seal ring on a vertical plane  The center of the seal ring is open to allow process system connection access to the ports on the top of the MCO  Rubber seal ring bladders are inflated by instrument air after the process hood-seal ring assembly is bolted into place  A cross-sectional view of the seal ring assembly is shown in Figure B2-14  Seal ring structural analysis was performed as discussed in SNF-4772, Evaluation of Cold Vacuum Drying Facility Seal Ring Loading and Stress

When installed and inflated, the seal rings provide the seal for the tempered annulus water at the top of the cask-MCO annulus and allows circulation of the tempered water to and from the annulus  The seal ring and process hood are bolted to the cask to resist the uplift load due to any credible water pressure from the circulating water within the cask-MCO annulus  The seal ring bladders are designed to withstand expected pressures generated by the tempered water (annulus) system  Various process connections, including the tempered water (annulus) system outlet connection and an instrument air connection, are provided

The following components of the tempered water (annulus) system are designated as safety-class

- MCO seal ring assembly tempered water outlet port with refill connection and gas vent port (inflatable seals are general service)

- Antisiphon valves

- Cask tempered water return (outlet) piping from the connection of the flexible lines to the support stand, including the portion of the line on the support stand that incorporates the antisiphon valves and connects to the seal ring outlet
- Cask tempered water supply piping from the connection of the flexible lines on the support stand to the lower cask port, including the lower cask port connection, quick-disconnect fitting, and outer flange connection and tubing

- Cask-MCO annulus inlet and outlet water low-level switches and combined water level gauges

- Cask-MCO annulus inlet water high temperature switches

- Tempered water level check petcock valves on supply piping (backup to level indicator and remote level alarm)

Both the tempered water supply and return lines have safety-class level and temperature switches and safety-class level gauges. If the level of the annulus water drops below a designated level, the low-level switches send a signal to the SCIC system (see Section B4 3 1) that initiates a safety-class alarm to signal the operators that the water level has dropped below the setpoint for the annulus. When the alarm is activated, corrective measures are taken by the operator. If refill deionized water cannot be added through the tempered water (annulus) system general-service recirculation loop, the operator can refill the annulus manually with water, using a funnel, through a fill port and valve in the outlet line attached to the cask-MCO seal ring. A vent port is provided on the seal ring to assist with the manual refill process. A level check petcock is provided on the supply piping to facilitate manual refill operation, by providing a visual indication that the annulus water level is above the level of the fuel inside the MCO. This SSC allows backup verification of an adequate cask-MCO annulus water level, if the safety-class level gauges with remote alarms cannot be reset because of leakage from the general-service seals on the process hood seal ring assembly. Note that the seal ring is significantly above the fuel level inside the MCO but below the safety-class level gauges. Two parallel safety-class antisiphon valves are connected between the safety-class portion of the inlet and outlet piping, to prevent water from being inadvertently siphoned from the annulus in the event there is a line break in the general-service piping.

The CVDF process requires that the MCO normal drying operation and dryness testing be performed between 40 °C to 50 °C. If the inlet water temperature registers high, the safety-class temperature switches send a signal to the SCIC system (see Section B4 3 1), which de-energizes the tempered water (annulus) system heater. Calculations verify that the stagnant annulus water will cool even when the decay heat of the fuel is taken into account. Tempered water temperatures are monitored via the MCS system. Low tempered water temperature during drying would be an abnormal condition and corrective actions would be in accordance with operating procedures.

**B4 3 4 3 Functional Requirements** The functional requirements for the tempered water (annulus) system safety-class components are as follows

- **Heat Transfer** — Provide heat transfer from the MCO to the cask's inner surface to accomplish the system function of removing decay and reaction heat from the fuel
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- **Low Annulus Level Alarm** — Detect low annulus water level and transmit a signal to the SCIC system, which will sound a safety-class alarm that alerts operators to the low annulus water condition.

- **Supply Piping Leak Detection** — Provide a means for detecting a leak in the tempered water (annulus) system supply pipe to the MCO. Maintain capability for the operator to visually check the annulus water level and to observe possible leakage conditions during MCO processing and response to SCIC low-level alarm conditions.

- **Emergency Annulus Refill** — Provide emergency annulus water refill capability if the general-service deionized water addition in the tempered water (annulus) system is not functioning and there is a loss of annulus water.

- **Antisiphon Design** — Prevent siphoning of water from the cask-MCO annulus during MCO processing in the event of a line break accident.

- **Temperature Detection** — Detect MCO inlet water high temperature (beyond the setpoint for a specific processing step) and, in conjunction with the SCIC system, shut down the tempered water (annulus) system water heater.

- **System Operability, Calibration, and Testing** — Ascertaining that the tempered water (annulus) system is operable. Safety-class components are functioning as required and setpoints for all safety-class level and temperature switches are as specified.

- **Single-Failure Evaluation** — Ensure the safety-class portions of the tempered water (annulus) system are single-failure proof.

- **Seismic Qualification** — For the safety-class portions, provide a leaktight tempered water (annulus) system and passively prevent loss of water from the cask-MCO annulus by use of seismically qualified components, piping, and antisiphon valves.

**B4 3 4 4 System Evaluation** The tempered water (annulus) system components that are identified as safety class will maintain the water level in the annulus and prevent overheating of the annulus water to prevent the accident scenarios described in Chapter B3 0. Line breakage or valve failure may occur in other general-service portions of the tempered water (annulus) system, without impacting the annulus water level to the extent that a thermal runaway reaction, MCO overpressurization, or internal hydrogen explosion accident could be initiated.

The tempered water (annulus) system safety-class components have been designed to meet and perform their functional and safety requirements. The system evaluations of each of the functional requirements are discussed separately below.
**Heat Transfer**  The massive metal structure of the MCO cask, in conjunction with the MCO, provides an excellent heat conduction path for the fuel material contained within the MCO. Heat removal from the fuel material is accomplished by use of the tempered water (annulus) system, which circulates water through the cask-MCO annulus. Design of the tempered water (annulus) system ensures that the annulus water level will not fall below a level that is at least 1 ft above the level of the fuel in the MCO. Calculations have been made to show that transfer of heat from the MCO to the tempered water (annulus) system is sufficient to prevent or mitigate the thermal runaway accident. Engineering calculations demonstrating the adequacy of the heat transfer performance are included in SNF-2770, *Cold Vacuum Drying Facility Design Basis Accident Analysis Documentation*. Experimental heat transfer testing is documented in HNF-4057, *Cold Vacuum Drying Proof of Performance (First Article Testing) Test Results*. Verification that the heat transfer codes used for modeling are consistent with experimental results is provided in SNF-4083, *Hanford SNF Cold Vacuum Drying Process Post Test GOTH SNF Model Simulations of Proof of Performance or First Article Test and Equipment Performance Evaluations*.

**Low Annulus Level Alarm**  A low annulus water level is detected by the redundant safety-class low-level switches, which are single-failure proof (SNF-4290) and fail safe on loss of electrical power according to vendor information. These switches are set for at least 1 ft above the fuel level in the MCO to activate the SCIC system to sound a safety-class alarm that must be acknowledged by the operators. Operators must take action to bring the annulus water level above the low-level switch setpoint before the alarm can be reset. The safety requirement is to maintain the annulus water level above the height of the fuel in the MCO.

**Supply Piping Leak Detection**  The tempered water (annulus) system safety-class supply line connects to the lower port of the MCO cask through a safety-class, quick-disconnect fitting. A safety-class, double-walled design is provided up to a height above the top of the fuel inside the MCO. This design configuration leaves the outer pipe open at the top so that a leak in the inner supply line connected to the cask is captured and water flows upward and out the open end. The cask-MCO annulus water level is normally maintained above the fuel level in the MCO, however, if a leak in the inlet line occurs, it would be detected by the annulus inlet and/or outlet low-level switches, and an SCIC system alarm would be activated. This leak could be initially detected by the non-safety-class MCS low-level alarm in the tempered water receiver tank and by operator visual inspection. To ensure there is no leak in the inner supply line when MCO processing begins, a leak check is performed on the cask tempered water inlet line connection.

Another provision for monitoring the cask-MCO annulus water level during MCO processing is provided by the safety-class level gauges. These gauges (combined with the low-level switches) will indicate a water leakage condition. The gauges are placed on the process hood support stand in clear view of the operator, so that they have the capability to alert the operator to abnormal conditions. These level gauges are also utilized during responses to low-level alarms. A level check petcock also allows visual verification that the cask-MCO annulus water level is above the level of the fuel in the MCO.
Emergency Annulus Refill  In emergency situations where the water level is low and the deionized water addition in the general-service portion of the tempered water (annulus) system is unavailable, deionized water can be manually added to the annulus through a refill connection (stub-out) on the seal ring outlet line. The operator adds deionized water from a 55-gal drum with hand truck using a hand pump and gallon bucket by pouring through a funnel at the stub-out until there is visible leakage out the vent port. This response requires that there is no leakage of water past the seal ring assembly seals. If there is leakage from the tempered water inlet line lower connection to the cask, the line can be disconnected and the cover plate installed to bring the MCO into a safe-standby condition. About 27 gal of water is required to fill the cask-MCO annulus to above the height of the fuel in the MCO. As a minimum, 30 gal of deionized water must be available to have a sufficient amount of water to fill the cask-MCO annulus and allow for spillage and leakage.

Antisiphon Design  There are two antisiphon bypass lines in the safety-class portion of the tempered water (annulus) system. The flow through each line is controlled by safety-class flow control valves. The design of this system ensures that a line break in the general-service portion of the tempered water (annulus) system will not lead to siphoning of the water from the cask-MCO annulus. The two bypass flow control valves each route 1 gal/min to the cask-MCO outlet line and break the pressure path between the inlet and outlet lines that otherwise could result in siphoning of water from the annulus if a line break occurs in the general-service portion of the tempered water (annulus) system.

Temperature Detection  Detection of inlet water high temperature is ensured through use of safety-class temperature switches that signal the SCIC system to shut off power to the tempered water (annulus) system heater. The trip setpoint of the SCIC system takes into consideration the error associated with the instruments and shuts off the tempered water heater prior to exceeding the parameter limit of 50 °C. Water must be cooled to below the setpoint temperature (see Table B4-6) before the SCIC system can be reset.

System Operability, Calibration, and Testing  All tempered water (annulus) system equipment must be periodically calibrated according to the respective manufacturer’s recommended schedule and practice. Calibration and test connections have been provided to enable in-service testing and calibration where practical. Modular design has been incorporated into the tempered water (annulus) system equipment to permit easy change-out of systems requiring timely repair and/or special skills, and to reduce problems associated with equipment removal and repair.

Single-Failure Evaluation  The safety-class portion of the tempered water (annulus) system has been evaluated by single-failure analysis in SNF-4290. A single random failure can be tolerated using operator response actions. The tempered water (annulus) system design incorporates a cask-MCO annulus manual refill station for recovery under process upset conditions. Recovery action can be accomplished in much less time than the approximately 22 hours determined to be available based on the accident analysis (SNF-4290).
protection of the supply line is provided by prohibiting any movement of the overhead crane that could result in a single accident that could fail the double-walled supply piping.

**Seismic Qualification** All safety-class components are designed to performance category 3 standards. System leakage is prevented by use of safety-class piping and cask and seal ring assembly port connections in key portions of the tempered water (annulus) system. Other components that are safety-class include level and temperature switches, level gauges, and redundant antisiphon valves. All of these tempered water (annulus) system components have been seismically qualified. Also, the annulus water supply line connecting to the port at the bottom of the cask is double-walled construction, up to a level exceeding the level of the fuel in the MCO, and is connected to the process hood support stand. To ensure performance under all required conditions, the process hood support stand that supports the tempered water (annulus) system safety-class equipment has a category 3 designation. If inadvertent system leakage should occur during processing, a sump with a general-service level alarm in the floor below the cask will collect the water and send a signal to the MCS. This design provision provides a defense-in-depth capability for water leakage from the cask-MCO annulus. Engineering calculations demonstrating the adequacy of the performance category 3 structural design are listed in SNF-3001.

**B4 3 4 5 Controls (Technical Safety Requirements)** The following assumptions associated with the tempered water (annulus) system require TSRs to ensure the performance of the safety functions:

- The SCIC tempered water temperature trip system is operable whenever an MCO is connected to the tempered water (annulus) system.
- The annulus inlet and outlet line low water level switches are operable prior to heatup or draining whenever an MCO is connected to the tempered water (annulus) system.
- The annulus inlet line high water temperature switches are operable whenever an MCO is connected to the tempered water (annulus) system.
- The annulus water level gauges are operable whenever an MCO is connected to the tempered water (annulus) system.
- A functional leak check has been performed on the supply line to the lower port of the MCO cask prior to heatup or draining.
- The cask-MCO annulus low water level switches, high water temperature switches, and water level gauges are calibrated and have correct setpoints before an MCO is connected to the tempered water (annulus) system.
Approximately 30 gal of deionized water and support equipment for manually adding deionized water to the cask-MCO annulus is available in a bay for operators before MCO process operations are initiated and during processing in that bay.

### B4 3.5 Vacuum Purge System Safety-Class Components

#### B4 3.5.1 Safety Function

The VPS provides safety-class process pressure information to the SCIC system that initiates an ISO & PURGE which de-energizes the VPS fail-closed safety-class isolation valves and is credited in the Chapter B3.0 accident analysis with prevention functions for the thermal runaway reaction. The VPS is credited with safety-class functions for the thermal runaway reaction and MCO overpressurization accidents and safety-significant functions for the internal hydrogen and external hydrogen explosion accidents. Safety functions performed by the VPS for these accidents are as follows:

- **MCO overpressurization** — Safety-class functions of the VPS are to provide isolation of the MCO from the VPS including deionized water and to make a backup 30 lb/in² gauge vent path available that precludes high pressures in the MCO. The VPS isolation valves, piping, and MCO connectors form a leaktight portion of the extended MCO pressure boundary for normal operation and when an ISO & PURGE occurs. Pressure transmitters which are components of the VPS, provide signals to the SCIC system for monitoring the MCO pressure during operation sequences and activating an ISO & PURGE if setpoints are exceeded. Safety-class filters on the instrument air lines to the actuators on the isolation valves ensure functional operation of the valves.

- **Thermal runaway reaction** — Safety-class functions of the VPS are to provide isolation of the MCO from the VPS, including deionized water through use of VPS isolation valves, piping, and MCO connectors, and to form a leaktight portion of the extended pressure boundary for the MCO for normal operation and when an ISO & PURGE occurs. Pressure transmitters which are components of the VPS, provide signals to the SCIC system for monitoring the MCO pressure during operation sequences and activating an ISO & PURGE if setpoints are exceeded. Safety-class filters on the instrument air lines to the actuators on the isolation valves ensure functional operation of the valves.

- **Internal hydrogen explosion** — The VPS performs all of the functions required for the thermal runaway reaction to prevent or mitigate the safety-significant consequences of the internal hydrogen explosion.

- **External hydrogen explosion** — This accident has safety-significant consequences. The VPS performs the same functions as the MCO overpressurization accident except the pressure monitoring is not a credited safety function.
The VPS plays a major role during processing by providing additional "water isolation" of the MCO from potential water ingress sources during and after the MCO processing steps involving proof-of-dryness demonstration. Redundant safety-class deionized water isolation valves prevent water from being introduced into the MCO after the pressure hold tests have been verified. Safety-class VPS pressure instrumentation is also used during the MCO pressure hold tests.

The VPS is designated general service, but certain VPS components are designated safety-class as discussed below.

**B4 3 5 2 System Description**

One-inch VPS lines are connected to each of two MCO ports. One of the VPS lines connects to the long process tube port. This line connects to the general-service helium system and the deionized water line and also has connections to the SCHe system. The other line (MCO filtered process exit port) provides a pathway directly to the general-service VPS equipment and also has connections to the SCHe system. In general, the VPS consists of isolation valves, piping, and instrumentation in conjunction with a residual gas analyzer, a condenser (which can be bypassed after most of the water is removed from the MCO), a condenser water collection tank routed to the PWC system, a helium line to provide pressurization to remove MCO water, and a four-stage roots vacuum pump that connects to the process vent. When bulk water has been drained from the MCO, a valve to the vacuum pump is opened and pump-down is initiated to remove the remaining water in the MCO. At various stages, the water content of the MCO is monitored by checking the temperature difference across the condenser. The condenser captures the majority of the water vapor before it reaches the vacuum pump. At some point, the monitoring gauges and the condenser delta temperature readings will indicate most of the water has been removed, and pumping is continued with the condenser bypassed. Process steps involving pressure test rise are used to verify the MCO is ready for shipment. More detailed VPS information is provided in Figure B2-17 and in SNF-3062, *SNF Project Cold Vacuum Drying Facility Vacuum Purge System Design Description*.

The only components of the VPS that are safety-class are those that isolate the MCO from the VPS, monitor pressure, and provide an auxiliary vent path in situations where the MCO pressure rise is greater than can be handled by the SCHe system vent. The following VPS components are designated safety class:

- MCO process connectors and flexible piping located between the MCO process ports and the process hood support stand.
- Hard piping from the MCO flexible hose connections to just beyond the second isolation valve in each MCO process port line where the lines connect to the flexible tubing from the PES.
- Fail-closed MCO isolation valves with filters on the instrument air line to the valve actuators (mounted on the process hood support stand).
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- Fail-closed deionized water isolation valves
- Pressure transmitters, pressure indicators, and flow controls (including instrument air line filters) mounted on the process hood support stand
- MCO 30 lb/in² gauge vent path to local exhaust duct

All of the above components are safety-class and are designed to performance category 3 standards. The safety-class pressure transmitters send signals to the SCIC system that activate an ISO & PURGE if process upset situations are encountered. The SCIC system will initiate an ISO & PURGE that will de-energize the fail-closed VPS and deionized water isolation valves, in conjunction with the PWC system and general-service helium system isolation valves, and begin an SCHe purge of the MCO. The 30 lb/in² gauge vent path is backup to the SCHe system vent path to ensure the pressure will not reach the MCO 150 lb/in² gauge rupture disk rating.

**B4 353 Functional Requirements** Functional requirements for the VPS safety-class components to perform their safety functions are as follows:

- **VPS Isolation** — Provide fail-closed VPS isolation valves (with instrument air line filters) to isolate the MCO from the non-safety class portion of the VPS and maintain the MCO extended pressure boundary
- **Deionized Water Isolation** — Provide fail-closed deionized water isolation valves (with instrument air line filters) to prevent water ingress and maintain the MCO pressure boundary
- **Process Piping** — Provide leak-tight process piping and flexible connection hoses from the MCO ports to the isolation valves for the MCO isolation pressure boundary
- **Connection Capability** — Provide leak-tight process connectors for joining flexible hoses to MCO processing ports
- **Pressure Detection** — Detect MCO pressure and provide a signal to the SCIC system to initiate MCO isolation and SCHe actuation during process upset conditions
- **Vent Path** — Provide a 30 lb/in² gauge vent path to the local exhaust duct
- **System Operability, Calibration, and Testing** — Ensure the VPS is operable and safety-class components are functioning as required and the pressure readings are reliable
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- **Single-Failure Evaluation** — Ensure safety-class portions of the VPS are single-failure proof
- **Seismic Qualification** — Provide leaktight VPS system (safety-class portion) by use of seismically qualified valves, sensors, piping, hoses, and other components

**B4 3 5 4 System Evaluation**  
The VPS safety-class components are required to isolate the MCO and to provide signal input of process parameters to the SCIC system. Each of the function requirements is discussed separately below

**VPS Isolation**  
The VPS line connected to the MCO filtered process exit port connector includes two electropneumatically operated safety-class isolation valves mounted to the process hood support stand. The 1-in., stainless steel, gas-operated valves are rated 150 lb/in² gauge. During normal drying operations, the valves are held open by instrument air. Following an SCIC trip condition that results in an ISO & PURGE, the VPS isolation valves move to the closed position (until reset) to establish and maintain this portion of the pressure boundary required for the SCHe system safety function. The valves also provide local position indication.

The isolation valves are a ball-type design that use a cylinder and piston air actuator to rotate the ball shaft. Each safety-class isolation valve is controlled by an associated safety-class electric solenoid valve. Normally, the solenoid valve is energized in the open position allowing instrument air supply to the VPS isolation valve actuator to hold the pistons in an open position. Under SCIC trip conditions, the SCIC de-energizes the solenoid valve that in turn closes the solenoid valve, blocks the instrument air flow, and allows existing air in the inner chamber to vent, which then causes the isolation valve to close via a spring closure mechanism. The solenoid valve operation is fail-safe in the respect that a loss of electrical power or loss of instrument air would cause the desired safety function to occur (i.e., the isolation valve would close). Safety-class filters are required on the instrument air lines to the actuators on the isolation valves to prevent buildup of foreign material and preclude potential valve malfunction. A general-service instrument air dryer eliminates concerns for moisture in the instrument air line.

The VPS also contains two safety-class gauge root valves, located on instrument tap lines, between the VPS hard piping and the VPS safety-class instrumentation. These valves are required to be in the open position for the VPS safety-class instrumentation to perform its functions, and they are rated at 150 lb/in² gauge internal pressure.

**Deionized Water Isolation**  
The VPS line from the MCO long process tube also contains two electropneumatically operated safety-class isolation valves. During flushing of the PWC system line, the valves are held open by instrument air but otherwise remain closed. Following an SCIC trip condition requiring SCHe actuation, the isolation valves move to the closed position to establish and maintain a portion of the pressure boundary required for the SCHe system safety function.
Each safety-class isolation valve is controlled by an associated safety-class electric solenoid valve. For flushing operations, the solenoid valve is energized in the open position allowing instrument air supply to the VPS isolation valve actuator. Under closed or SCIC trip conditions, the SCIC de-energizes the solenoid valve, closing the solenoid valve and blocking the instrument air flow, which then causes the isolation valve to close via a spring closure mechanism. The solenoid valve operation is fail-safe in the respect that a loss of electrical power or loss of instrument air would cause the desired safety function to occur (i.e., the isolation valve would close). Filters are also required on the deionized water valve instrument air lines.

**Process Piping** Safety-class flexible piping provides a flow path between the MCO process connectors and the VPS hard piping mounted on the process hood support stand. The flexible piping forms a portion of the primary confinement boundary during times when the MCO process plugs are open. The flexible pipe has high integrity end fittings that are connected to the process connectors and to the process hood support stand VPS piping. The flexible pipe is made of Teflon with an anti-kink cover and a protective metal armor and has a 1-in inner diameter with a design pressure of 150 lb/in² gauge.

The VPS also has safety-class hard piping on the process hood support stand. This piping runs from the connection with the flexible piping to the isolation valves and (in the case of the MCO vacuum port connection) then to the flange connector at the process hood support stand that connects to the PES. The 1-in piping is made of stainless steel and forms a portion of the MCO primary confinement boundary. The pipe is rated for 150 lb/in² gauge internal pressure.

**Connection Capability** Safety-class MCO process connectors provide a leak-tight confinement attachment between the MCO shield plug process ports and the flexible piping that connects to the VPS equipment installed on the process hood support stand. The process connectors also provide MCO connection points for the PWC tubing. The connectors are manually bolted to the MCO shield plug — one connector by five bolts and the other by four bolts — to ensure installation on the correct process port. Following installation, the mechanical seal connections must pass a pressure test before CVDF operations may proceed. The space between the double O-rings on the process connectors can be leak tested to verify the connectors are leak tight. Internal to the connectors is a socket mechanism that is used to open and close the MCO port plug. This mechanism is designed so the MCO plug cannot be removed from the port while the connector is in place. The connectors are specially designed for this application and are rated for an internal pressure of 150 lb/in² gauge.

**Pressure Detection** VPS safety-class instrumentation provides signals used by the SCIC system to initiate automatic protective actions. The protective actions, depending on the process upset detected by the instruments, involve multiple systems and components. For a more detailed description of SCIC protective actions, see Section B4 3.1.2. There are two different types of pressure transmitters. In addition, there are pressure indicators. One type of safety-class pressure transmitter measures pressure over the range of -14.7 lb/in² gauge to +12 lb/in² gauge for specific

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1Teflon is a trademark of Du Pont de Nemours & Company
MCO processing steps  The other type of pressure transmitter has a range of 0 to 100 torr and is used when the final stages of processing indicate most of the water has been removed from the MCO and the VPS is capable of reaching a vacuum below 12 torr. Both types of pressure transmitters are installed on the VPS hard piping between the connection from the flexible line to the MCO filtered process exit port and the VPS safety-class isolation valves in a redundant arrangement. Each transmitter signal is independently sent to the SCIC system.

Two safety-class local pressure indicators, which indicate MCO pressure, are also provided in the VPS. The redundant indicators are installed in the same location as the pressure transmitters. The indicators provide local indication only and are qualified to be operational after the DBE. All pressure instrumentation is rated at 150 lb/in² gauge internal pressure.

Vent Path The 30 lb/in² gauge vent path has been provided as backup in the remote possibility that the MCO pressure rise exceeds the capability of the SCHe system vent path. The vent system consists of a combination safety-class rupture disk and spring-type, safety-class check valve that will relieve pressure above 30 lb/in² gauge. The system is connected to the MCO filtered process exit port and vents into the local exhaust duct to ensure hydrogen coming from the MCO is diluted below the lower flammability limit. The 30 lb/in² gauge vent system is a redundant provision for protection of the MCO 150 lb/in² gauge rupture disk. Engineering calculations that support the design of the 30 lb/in² gauge vent path are provided in SNF-4023, *CVD External Hydrogen Management Evaluation*, and are listed in SNF-3001.

System Operability, Calibration, and Testing Safety-class VPS components are designed to be functionally tested for actuation given a simulated process condition input (see Section B4.3.1.4 for additional functional information). System operability is verified by surveillance of the system’s operability status and component states (e.g., valve positions and pressure readings) before enabling the system for each MCO process cycle. Additional periodic surveillance is required if the system is to be operated for an extended period beyond one normal processing cycle. Periodic testing of VPS safety-class components is dictated by the requirements of the individual components, according to the respective manufacturer’s recommended schedules and practice, and is administered by controlled procedures for all safety-class and safety-significant SSCs.

Recalibration of signals from the pressure transmitters to the SCIC system is maintained according to the respective manufacturer’s recommended schedule and practice. The VPS is designed with calibration and test connections to enable periodic in-service measurement and calibration of all setpoints and adjustments that affect the manner in which the VPS performs.

Single-Failure Evaluation The safety-class portion of the VPS has been evaluated by single-failure analysis in SNF-4290. This analysis shows that safety-class SSCs meet the single-failure criterion of DOE Order 6430.1A and a single random failure of the VPS safety-class components is tolerable. In general, operators will be aware of any system leakage because the system will not pump down to the required pressure, and action will be taken to eliminate the problem. The fact that there are two pressure transmitters of each type with different ranges.
allows a one-over-one check and SCIC system channel checks, train A to train B, that will immediately show where recalibration or pressure transmitter replacement is needed.

**Seismic Qualification** All safety-class components of the VPS required to perform the safety-class primary confinement functions are designed and qualified for performance category 3. The safety-class components, except for the flexible connection hoses, are mounted on the MCO or the process hood support stand, which are both seismically qualified for performance category 3. The flexible hoses in the VPS ensure there will be no line breakage during a seismic event.

**B4 3 5 5 Controls (Technical Safety Requirements)** The following assumptions associated with the VPS require TSRs to ensure performance of the safety functions:

- The VPS isolation valves have been tested and are operational prior to opening the MCO port valves whenever the VPS process port connectors are attached to the MCO.
- The deionized water isolation valves have been tested and are operational prior to opening the MCO port valves whenever the VPS process connectors are attached to the MCO.
- MCO process connectors are securely attached to the MCO and have been pressure-tested according to procedures.
- All pressure transmitters that provide signals to the SCIC system have been tested and are operational whenever the VPS process connectors are attached to the MCO.
- The pressure indicators have been calibrated and are functional prior to opening the MCO port valves whenever the VPS process connectors are attached to the MCO. The 30 lb/in² gauge vent line is connected to the filtered process exit port connector and is operational prior to processing whenever the VPS process connectors are attached to the MCO.

**B4 3 6 General-Service Helium System Safety-Class Components**

**B4 3 6 1 Safety Function** The general-service helium system has the overall safety functions of providing helium at a safe pressure with appropriate flow monitoring to different parts of the MCO processing systems, and providing fail-closed general-service helium system isolation valves that form part of the extended pressure boundary for the MCO in the event that process upset conditions are detected by the SCIC system and an ISO & PURGE occurs. Components of the general-service helium system have been analyzed in the Chapter B3 0 accident analyses to.
perform certain safety functions for four specific event scenarios: the thermal runaway reaction, MCO overpressurization, internal hydrogen explosion, and external hydrogen explosion accidents.

The general-service helium system components perform the following safety functions for these accidents:

- **MCO overpressurization** — Isolation of the MCO by the general-service helium system components is a safety-class function for this accident. Safety-class filters on the instrument air lines to the isolation valve actuators ensure functional operation. The general-service helium system safety-class safety relief valves also prevent an MCO overpressurization accident that could otherwise result from helium overpressurization of the MCO.

- **Thermal runaway reaction** — The general-service helium system components provide a safety-class function of providing isolation of the general-service helium line (connected to the VPS line to the MCO) when the safety-class valves are de-energized by the SCIC system and, in conjunction with the piping, form a portion of the extended pressure boundary for the MCO. Filters on the instrument air lines to the isolation valve actuators ensure functional operation.

- **Internal hydrogen explosion** — Isolation of the MCO by the general-service helium system components is a safety-significant function for this accident. In addition, the general-service helium system provides low helium flow signals to the SCIC system as a key process parameter to initiate an ISO & PURGE that prevents safety-significant consequences from the internal hydrogen explosion accident. Filters on the instrument air lines to the isolation valve actuators ensure functional operation.

- **External hydrogen explosion** — In this accident, isolation of the MCO by the general-service helium system is a safety-significant function. A portion of the extended pressure boundary of the MCO is provided by the general-service helium system isolation valves and piping.

The safety-class components in the general-service helium system have been designated as such because of their safety-class and safety-significant functions for process control and accident prevention. While the MCO is connected to the VPS, a safety-class confinement boundary is required during accident conditions. As a consequence, the MCO envelope is extended to include the isolation valves within the general-service helium system piping that connects to the VPS piping extending from the long process tube MCO port connector. The flow elements/flow-indicating transmitters that provide signals to the SCIC system also are designated safety class because they must provide reliable information under processing conditions to the SCIC system.

**B4 3 6 2 System Description** — The general-service helium system is located both inside and outside the CVDF. Up to two leased helium tube transporters are parked on a concrete pad located on the west side of the facility. A helium pipe penetrates the west wall and connects to a
helium header running north-south along the full length of the east wall of the transfer corridor. Branch piping connects each process bay to the helium header, and flexible piping is used to distribute the helium to the PES, the process hood, support stand, and VPS, and the SCHe system. Within the process hood support stand, the general-service helium is hard piped to the VPS piping connected to the MCO long process tube connector.

In addition to piping, the general-service helium system includes pressure regulators, pressure and flow indicators, flow control valves, and pressure relief valves. System control and alarm functions are provided by the MCS (see Section B2.5.9.1). Check valves are used to prevent backflow of one process bay from reaching the other process bays. The two tube transporters provide a minimum 2-week supply of helium at the maximum process demand rate. Continuous operation is provided by allowing an empty tube transporter to be disconnected and replaced while drawing helium from the other. More detailed general-service helium information is provided in Figures B2-17 and B2-20 and in SNF-3067, _SNF Project Cold Vacuum Drying Facility General Service Helium System Design Description_.

The safety-class portion of the general-service helium system is located within the process hood support stand and where the high pressure safety relief valves are placed on the line from the tube trailer. The following components of the general-service helium system are designated safety-class:

- Hard piping on the process hood support stand from the connection to the VPS line to the opposite end, where the piping connects to the flexible hose from the PES
- Redundant fail-closed isolation valves with associated instrument air filters
- Redundant pressure relief valves on the CVDF general-service helium supply line in the transfer corridor
- Redundant helium flow elements/flow-indicating transmitters located on the process hood support stand

The portion of the system between the MCO and the isolation valves forms part of the safety-class pressure boundary, when the MCO is isolated from the process systems by the SCIC system. The redundant flow elements/flow-indicating transmitters support the SCIC system by providing reliable helium flow information. Redundant safety relief valves protect against general-service helium system regulator failures that could cause MCO overpressurization.

**B4 3 6 3 Functional Requirements**

The functional requirements needed for the general-service helium system safety-class components to perform their safety functions are as follows:

- **General-Service Helium Isolation** — Provide fail-closed isolation valves (with instrument air line filters) to seal off the MCO from the general-service helium system normal inlet and maintain the MCO extended pressure boundary.
- **Process Piping** — Provide leaktight process piping to the VPS line (long process tube connection)

- **Helium Flow Detection** — In addition to ascertaining that helium flow to the MCO is sufficient, provide a low flow signal to the SCIC system that will initiate MCO isolation and activate the SCHe system if the helium flow is less than the setpoint

- **Helium Overpressurization Protection** — Ascertan that the general-service helium system main supply header's safety-class safety-relief valves are operational, set at the correct pressure, and are reliable

- **System Operability, Calibration, and Testing** — Ensure the general-service helium system is operable, safety-class components are functioning as required, and flow readings are reliable

- **Single-Failure Evaluation** — Ensure safety-class portions of the general-service helium system are single-failure proof

- **Seismic Qualification** — Provide leaktight general-service helium system (safety-class portion) by use of seismically qualified isolation valves and piping

**B4 3 6 4 System Evaluation** The general-service helium system safety-class components have been designed to meet and perform their functional and safety requirements. System evaluations of each of the functional requirements are discussed separately below.

**General-Service Helium Isolation** The general-service helium line has two electropneumatically operated safety-class isolation valves mounted to the process hood support stand and located in the piping that connects to the VPS line that runs to the MCO (long process tube) connector. These valves are identical to the ones used for isolating the VPS. The 1-in stainless steel, gas-operated valves are rated at 150 lb/in² gauge. During normal drying operations, the valves are held open by instrument air. Following an SCIC trip condition that initiates an ISO & PURGE, the general-service helium isolation valves move to the closed position (until reset) to establish and maintain a portion of the pressure boundary required for the SCHe system safety function. The valves are also provided with local position indication.

The isolation valves are a ball-type design that use a cylinder and piston air actuator to rotate the ball shaft. Each safety-class isolation valve is controlled by an associated safety-class electric solenoid valve. Normally, the solenoid valve is energized in the open position, allowing instrument air supplied to the general-service helium system isolation valve actuator to hold the pistons in an open position. Under SCIC trip conditions, the SCIC de-energizes the solenoid, which in turn, closes the solenoid valve, blocks the instrument air flow, and allows existing air in the inner chamber to vent out, which then causes the isolation valve to close via a spring closure mechanism. The solenoid valve operation is fail-safe in the respect that a loss of electrical power or loss of instrument air would cause the desired safety function to occur (i.e., the isolation valve...
would close) Safety-class filters are required on the instrument air lines to the actuators on the isolation valves to prevent buildup of foreign material and to preclude potential isolation valve malfunction. A dryer has been provided in the general-service instrument air system to eliminate concern for moisture buildup in the instrument air lines.

**Process Piping** Safety-class hard piping for the general-service helium system is located on the process hood support stand. At one end the hard pipe is connected to the flexible hoses from the PES, and at the other end the line joins the VPS hard piping coming from the MCO connector (long process tube). The piping is made of stainless steel and is 1-in in diameter. A major function of the piping is to form an extended primary confinement boundary for isolation of the MCO. The piping must be leak tight.

**Helium Flow Detection** Two safety-class flow element/flow-indicating transmitters are located in the hard piping line outside the MCO isolation valves and these components form the interface with the SCIC system for helium flow monitoring. These flow elements/flow-indicating transmitters are safety-class, however, they have a safety-significant function for the internal hydrogen explosion accident. The flow elements/flow-indicating transmitters are linked to the SCIC system mode switch and the required helium flow varies with each specific process step. If helium flow is not sufficient (above the setpoint) for a given process step, the SCIC system will initiate an ISO & PURGE.

Ascertaining that the general-service helium system flow is above the setpoint is necessary for the operators to be confident that the MCO processing steps are being carried out as required. Therefore these flow elements/flow-indicating transmitters also provide essential information remotely to the MCS for CVDF operations.

**Helium Overpressurization Protection** Two safety-class safety relief valves are provided in the 0.75-in stainless steel line connected to the leased tube trailers that supply helium. These safety relief valves are mounted in series and are set at 25 lb/in² gauge to ensure the MCO cannot be overpressurized in the unlikely event that the regulator and relief valves supplied with the tube trailers fail and the full pressure of the helium tanks is applied to the general-service helium line. Flow capacity of each redundant relief valve to preclude pressurization of the MCO above 150 lb/in² gauge is documented in calculation ME1-2621-ME-03, *Size the Helium Gas Piping for the Cold Vacuum Drying Facility*.

**System Operability, Calibration, and Testing** Two helium tube trailers are provided to ensure general-service helium system reliability. Both trailer tube systems can be connected to the general-service helium line but only one trailer is in use. The other acts as a standby supply. Surveillance and in-service inspections of the safety-class and safety-significant general-service helium system components are conducted in accordance with the manufacturer’s recommendations for respective components. These components are using administrative control for all testing, surveillance, and maintenance, and these activities are performed using controlled procedures. System operability is verified by surveillance of the system’s component states before.
enabling the system for each MCO process cycle. Additional surveillance is required if the system is to be operational beyond one normal processing cycle.

Safety-class general-service helium components that provide information to the SCIC system are designed to be functionally tested for operability using a simulated process condition input (see Section B4 3 1 4 for additional functional information). All safety-class and safety-significant components are calibrated according to the manufacturer's procedures and standards. Calibration and test connections are provided to enable in-service testing and calibration of equipment.

**Single-Failure Evaluation** The general-service helium system safety-class components have been evaluated by single-failure analysis (SNF-4290). A single random failure of the general-service helium system safety-class components has been shown to be tolerable. The non-safety class MCS also provides surveillance of the general-service helium system flow rates, which provides added assurance that any system malfunction will be detected early and responded to by operators.

**Seismic Qualification** The general-service helium system safety-class isolation valves and piping are seismically qualified and mounted to meet performance category 3 requirements. The isolation valves are electropneumatically operated valves that (1) fail closed upon loss of control power or instrument air, (2) ignore all other non-safety control signals and remain closed until reset by the SCIC system, and (3) provide reliable, leak-tight operation. These safety-class isolation valves are mounted on the process hood support stand, which is also seismically qualified for category 3 performance. The helium flow element/flow-indicating transmitters are not performance category-3 (seismically qualified) because they are not located within the safety-class isolation boundary and would be isolated from the MCO in a seismic event. The main supply header safety-class safety relief valves are seismically qualified and are in addition to a general-service safety relief valve provided on the helium tube trailers.

**B4 3 6 5 Controls (Technical Safety Requirements)** The following assumptions associated with the general-service helium system require TSRs to ensure performance of the safety functions:

1. The general-service helium isolation valves have been tested and are operational prior to opening the MCO port valves whenever the VPS process port connectors are attached to the MCO.
2. Flow elements/flow-indicating transmitters that provide signals to the SCIC system are functional prior to making process connections to the MCO.
3. The general-service helium line safety-class pressure relief valves are set at the correct pressure (25 lb/in² gauge) that will preclude MCO overpressurization.
Functional tests and leak testing of the isolation valves have been performed on an appropriate schedule

Calibration of the flow elements/flow-indicating transmitters has been performed on an appropriate schedule

**B4 3 7 Process Water Conditioning System Safety-Class Components**

**B4 3 7 1 Safety Function** The PWC system has an overall safety function of isolating the PWC system line to the MCO process port should process upsets occur. Components of the PWC system are analyzed in the Chapter B3 accident analyses to prevent or mitigate the safety consequences of a thermal runaway reaction, MCO overpressurization, external hydrogen explosion, internal hydrogen explosion, gaseous release, and liquid release.

The PWC system components perform the following safety functions for the above accidents:

- **MCO overpressurization** — PWC system isolation valves, instrument air line filters for isolation valves and piping are credited with a safety-class function of isolating the MCO for the MCO overpressurization accident.

- **Thermal runaway reaction** — The PWC system isolation valves and hard piping provide the safety-class function of isolating the PWC system line to the MCO and forming a portion of the extended pressure boundary for the MCO. Filters on the instrument air lines to the isolation valve actuators ensure functional operation.

- **External hydrogen explosion** — The PWC system isolation valves and hard piping provide the safety-class function of isolating the PWC system line to the MCO and forming a portion of the extended pressure boundary for the MCO. Filters on the instrument air lines to the isolation valve actuators ensure functional operation.

- **Internal hydrogen explosion** — PWC system isolation fulfills a safety-significant function for the internal hydrogen explosion.

- **Gaseous release** — Design and installation of the PWC drain line that runs through the process bays prevents safety-significant gaseous leaks into the bays that could result from corrosion and environmental conditions.

Components of the PWC system have been designated safety-class or safety-significant because they have safety-class or safety-significant functions for process control and accident prevention. The PWC system is connected to the MCO long process tube port, and during...
accident conditions, a safety-class confinement boundary must be established. When the isolation valves of the PWC system are closed, they become part of the extended isolation boundary for the MCO.

**B4 3 7 2 System Description** The PWC system is general service (except for the specific components listed in this subsection) and has two major purposes. First, the system receives bulk water from the MCO during draining operations and provides a vacuum source for aid in MCO water removal. Second, the PWC system conditions the MCO drain water by removing radioactive particulate and soluble species using filtration and ion exchange. The system is designed to service only one MCO from any one of the four bays at a time, so priority determination and scheduling are important. The PWC system vacuum source is a water jet ejector that requires continuous water circulation through a process loop arrangement. This jet ejector system has two water pumps (one for backup) to provide a vacuum for draining the MCOs, VPS condenser tanks, and the tempered water (annulus) system, or a cask-MCO annulus. Water is collected in receiver tanks, processed through an ion exchange module, and transferred to a storage tank to await return shipment to the K West Basin. The geometry of the tanks in the ion exchange modules provides defense-in-depth protection (as discussed in Chapter B6 0) to ensure that accumulation of a quantity of fissionable material that could result in an effective neutron multiplication factor ($k_{eff}$) of greater than 0.95 will not occur. Refer to Figure B2-22 and SNF-3082, *SNF Project Cold Vacuum Drying Facility Process Water Conditioning System Design Description*, for more detailed information.

The calculated operational limit for the mass of plutonium in an ion exchange module is 200 g each. The ion exchange modules are removed from service before they accumulate 80 nCi/g of transuranic material even if they have used only a fraction of their ion removal capacity. This is an operational convenience that allows the ion exchange modules to be economically disposed of as low-level waste. The criticality limit for loading of the ion exchange modules is identified in Section B6 3 5 as 875 kg of 1 25 wt% enriched UO$_2$.

An intact MCO pressure boundary is provided, in part, by the piping and the redundant isolation valves that will fail closed to isolate the MCO from the PWC. The SCIC system seismic detector is interlocked to the two water pumps, so the pumps will be de-energized in a seismic event. This interlock is not credited with a safety function and is general service.

The safety-class MCO isolation portion of the general-service PWC system is located within the process hood support stand. The components of the PWC system that are designated safety-class are as follows:

- Redundant fail-closed isolation valves with associated instrument air filters
- All piping, from the flexible piping and connection to the MCO long process tube port connector, to and including the hard piping that connects with the CVDF transfer line
The PWC drain line is a single pipe header that runs through all bays to the process water tank room for MCO free water draining. This pipe is safety-significant for confinement.

**B4 3 7 3 Functional Requirements** The functional requirements needed for the PWC system safety-class and safety-significant components to perform their safety functions are as follows:

- **PWC System Isolation** — Provide fail-closed isolation valves (with instrument air line filters) to seal off the MCO from the PWC system and maintain the MCO extended pressure boundary.

- **Process Piping** — Provide leaktight process piping from the connection with the facility PWC transfer line to the MCO long process tube connector port of the VPS.

- **PWC Transfer Line** — The PWC system transfer line to the receiver tanks shall be designed and installed to prevent random leaks caused by corrosion or environmental conditions.

- **System Operability, Calibration, and Testing** — Ensure the PWC system is operable and safety-class and safety-significant components are functioning as required.

- **Single-Failure Evaluation** — Ensure safety-class portions of the PWC system are single-failure proof.

- **Seismic Qualification** — Provide leaktight PWC system (safety-class portion) by use of seismically qualified valves, interlock trips, and piping.

**B4 3 7 4 System Evaluation** The safety-class features of the PWC system are accomplished by components that have been designed to meet and perform their functional and safety requirements. Each of the functional requirements are discussed separately below.

**PWC System Isolation** The PWC system line from the MCO (long process tube) connector has two electropneumatically operated safety-class isolation valves mounted to the process hood support stand. The 1-in stainless steel gas-operated valves are rated 150 lb/in² gauge. During the normal MCO draining and flushing operations, the valves are held open by instrument air. Following an SCIC trip condition that results in an ISO & PURGE, the PWC system isolation valves lock in the closed position (until reset) to establish and maintain the pressure boundary required for the SCHe system safety function. The valves also provide local position indication.

The isolation valves are a ball-type design with Teflon seats to accommodate particulate carried over from the MCO. These valves use a cylinder and piston air actuator to rotate the ball shaft. Each safety-class isolation valve is controlled by an associated safety-class electronic solenoid valve. Normally, the solenoid valve is energized in the open position, which allows...
instrument air supplied to the PWC system isolation valve actuator to hold the pistons in an open position. Under SCIC trip conditions, the SCIC de-energizes the solenoid, which in turn closes the solenoid valve, blocks the instrument air flow, and allows existing air in the inner chamber to vent out, which then causes the isolation valve to close via a spring closure mechanism. The solenoid valve operation is fail-safe in that a loss of electrical power or loss of instrument air would cause the desired safety function to occur (i.e., the isolation valve would close). Safety-class filters are required on the instrument air lines to the actuators on the isolation valves to prevent buildup of foreign material and to preclude potential isolation valve malfunction. A dryer has been provided in the general-service instrument air system to eliminate moisture buildup in the instrument air lines.

**Process Piping** Safety-class piping for the PWC system is located on the process hood support stand and includes both hard and flexible piping that connects with the MCO long process tube connector. At one end, hard piping (with isolation valves in the line) is connected to the hard piping from the CVDF PWC system transfer line. At the other end, the hard piping connects to flexible piping that is connected to the MCO port connector. The hard piping is made of stainless steel and has a 1-in inside diameter. The flexible pipe is made of Teflon, with an anti-kink cover and a metallic armor, and has a 1-in inner diameter with a design pressure of 150 lb/in² gauge. A major function of the piping is to form an extended primary confinement boundary for isolation of the MCO.

**PWC Transfer Line** A 1-in stainless steel transfer line runs from each of the bay process hood stations to the ejector on the PWC system. The line is mounted 23 ft above the floor, and installation is designed and secured to preclude damage from fretting and/or expansion. The use of stainless steel ensures there will be no corrosion over the design life of the facility (5 years). All line joining is conducted according to standard procedures for stainless steel that are proven by a long history of working with this material. The transfer lines are inspected for leakage after joining.

**System Operability, Calibration, and Testing** For operations, the PWC system has two water recirculation pumps to provide confidence that the system will be available for MCO processing. Periodic testing of the PWC system safety-class and safety-significant components is dictated by the requirements of the individual components, according to the manufacturer's recommended schedule and practice. The safety-class and safety-significant components of the PWC system are under administrative control for all testing, surveillance, and maintenance, and these activities are performed using controlled procedures. Safety-class PWC system components that provide information to the SCIC system are designed to be functionally tested for activation using a simulated process condition input (see Section B4 3 1 4 for additional functional information).

**Single-Failure Evaluation** The PWC system safety-class components have been evaluated by single-failure analysis in SNF-4290. A single random failure of the PWC system safety-class components has been shown to be tolerable.
Seismic Qualification  The PWC system safety-class components and piping are rated above the required pressure and are seismically qualified and mounted to meet performance category 3 requirements. The isolation valves that are electropneumatically operated (1) fail closed upon loss of control power or instrument air (2) ignore all other nonsafety control signals and remain closed until reset by the SCIC system and (3) provide reliable leaktight operation. The safety-class isolation valves and piping are mounted on the process hood support stand, which is also performance category 3 seismically qualified.

The safety-significant PWC system transfer tubing is mounted on the safety-significant CVDF building structure. The piping and mounting are in accordance with category 2 seismic performance category and is a credited design feature.

B4 3 7 5 Controls (Technical Safety Requirements)  The following assumptions associated with the PWC system require TSRs to ensure performance of the safety functions:

- Safety-class PWC isolation valves have been tested and are operational prior to opening the MCO port valves whenever the VPS port connectors are attached to the MCO.

B4 4 SAFETY-SIGNIFICANT STRUCTURES, SYSTEMS, AND COMPONENTS

As identified in Chapter B3, many of the safety-class SSCs also perform safety-significant functions. Those SSCs, including their safety-class and safety-significant functions, are discussed in Section B4 3 and are not repeated here. The SSCs credited with only a safety-significant function in Chapter B3 0 and detailed in this section are listed below:

- Building features
- Process general supply/exhaust HVAC system (HVACD system) components
- Process bay local exhaust HVAC and process vent system (HVACC/PV system)
- Process bay recirculation HVAC system (HVACB system) components
- Reference air system components
- Instrument air system components
- Standby power system
- Specialty equipment and special tools

Safety-significant SSCs were designed and fabricated in accordance with the required codes and standards identified in Table B4-3. Safety-significant SSCs were designed, fabricated and procured to the same quality assurance program as safety-class SSCs (i.e., use of an approved vendor or specific dedication). Adequate specification of codes and standards exists for safety-significant SSCs, either specific identification of items (e.g., valves and instruments) via data sheets, code and standard specification for design, fabrication, and construction (e.g., bulk commodities) or specific design (cold vacuum drying specialty items). Engineered safety significant items were designed in the same fashion as safety-class items. Seismic qualification for...
safety-significant items to performance category 2 is adequate to demonstrate required functionality. All systems and components are subject to construction and performance testing subsequent to installation in the CVDF.

Safety-significant SSCs that must retain pressure boundary/structural integrity during or after a seismic event were designed for performance category 2 seismic loadings. These designs were performed in accordance with seismic accelerations specified in HNF-PRO-097, Engineering Design and Evaluation. There is no active safety-significant equipment required to operate after a seismic event. Anchorage of performance category 2 SSCs was designed to the requirements of WHC-SD-GN-DGS-30006, Seismic Design Guide for Safety Class 3 and 4 Equipment at the Hanford Site.

B4-41 Building Features

B4-41.1 Safety Function. The CVDF structure is analyzed in Chapter B3.0 to perform the following safety functions to mitigate safety-significant consequences from the bounding accident scenarios:

- The process bay area structure (process bays 2, 3, 4 and 5) and process water tank room structure provide a confinement function in conjunction with the process general supply and exhaust HVAC system and the process bay local exhaust HVAC and process vent system. The physical structure allows for confinement of radioactive material released in the process bay during processing operations and in the process water tank room during PWC pumping. Its passive presence allows for mitigation of the radioactive release from within the CVDF after the gaseous release, liquid release, external hydrogen explosion, internal hydrogen explosion, and MCO overpressurization accidents, as indicated in Table B4-8, by allowing the ventilation systems to maintain a negative pressure in the process bays and process water tank room during facility operations as indicated.

The CVDF process building also is assumed in Chapter B3.0 to perform the following safety functions in support of safety-class systems:

- The CVDF process bay structures provide performance category 3 resistance to natural phenomena hazard events to protect the process equipment and to accomplish 3-over-1 seismic qualification to preclude damage or impairment of safety-class systems or components. Compliance with 3-over-1 requirements is documented in HNF-4291, Seismic Adequacy Review of PC012 SCEs that are Potential Seismic Hazards with PC3 SCEs — CVF Facility. Of specific concern for 3-over-1 protection is the process hood support stand and its flexible piping connections to the process connectors attached to the MCO. The SCHe and the SCIC systems must also be protected from damage. These systems and components provide preventive or mitigative functions for the accident conditions. Protection of the safety-significant SSCs is also provided.
Table B4-8 Confinement Function of Structures in Conjunction with Safety-Significant Heating Ventilation and Air Conditioning Systems During Design Basis Accident Conditions

<table>
<thead>
<tr>
<th>Design basis accident</th>
<th>Structure</th>
<th>HVAC system</th>
<th>Processing mode</th>
<th>SS DP alarm to control room</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gaseous release</td>
<td>Process bays 2 3 4 5</td>
<td>HVAC/PV and HVACD</td>
<td>MCO processing</td>
<td>Reference air</td>
</tr>
<tr>
<td></td>
<td>Process water tank room</td>
<td>HVACD</td>
<td>PWC pumping</td>
<td>Reference air</td>
</tr>
<tr>
<td>Liquid release</td>
<td>Process water tank room</td>
<td>HVACD</td>
<td>PWC pumping</td>
<td>Reference air</td>
</tr>
<tr>
<td>MCO external hydrogen explosion</td>
<td>Process bays 2 3 4 5</td>
<td>HVAC/PV and HVACD</td>
<td>MCO processing</td>
<td>Reference air</td>
</tr>
<tr>
<td>MCO internal hydrogen explosion</td>
<td>Process bays 2 3 4 5</td>
<td>HVAC/PV and HVACD</td>
<td>MCO processing</td>
<td>Reference air</td>
</tr>
<tr>
<td>Thermal runaway reaction</td>
<td>NA</td>
<td>HVAC/PV</td>
<td>MCO processing</td>
<td>NA</td>
</tr>
<tr>
<td>MCO overpressurization</td>
<td>Process bays 2 3 4 5</td>
<td>HVAC/PV and HVACD</td>
<td>MCO processing</td>
<td>Reference air</td>
</tr>
</tbody>
</table>

DP = differential pressure  
HVAC = heating ventilation and air conditioning  
HVAC/PV = process bay local exhaust HVAC and process vent (system)  
HVACD = process general supply/exhaust HVAC (system)  
MCO = multi canister overpack  
NA = not applicable  
PWC = process water conditioning  
SS = safety significant

- The CVDF structure provides seismically qualified mounting locations for safety-class items. The process hood support stand and SCHe support racks are anchored to the floor.
- The CVDF structure supports the process bay bridge cranes which are considered important-to-safety Category C. These cranes are considered part of the building features.

B4 4 1 2 System Description The CVDF process building is classified as a nonreactor nuclear facility according to DOE Order 6430 1 A, Section 1300. The office area is a nonnuclear facility rated for office or business use in accordance with the Uniform Building Code (UBC) (ICBO 1994). The CVDF is a temporary structure with a processing design life of 5 years. The building code requirements are met as defined in the UBC (ICBO 1994) for a Group H-7 occupancy for the process bay and support areas and a Group B occupancy for the administrative office areas.

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The CVDF building is an above-grade structure with approximate overall dimensions of 230 ft long by 80 ft wide. There are four significant sections to the CVDF structure: process bays, transfer corridor, process water tank room, and administrative areas.

The process bay area (60 ft by 150 ft) consists of four process bays plus one unused bay, with an adjoining process water tank room (20 ft by 35 ft). The unused bay is not outfitted with cold vacuum drying equipment or mezzanines, but does have the SCIC seismic detector electrical panels mounted on the south wall. These areas are constructed as a single-story, steel-frame building with an exterior wall of precast concrete panels. The process support areas (20 ft by 150 ft) include the transfer corridor and adjacent rooms along with the second floor mechanical room. This area is constructed as a two-story, steel-frame building with an exterior wall of metal siding. The administrative area is an adjacent single-story, pre-engineered metal building with exterior skin of insulated metal panels. All roof decks are metal. The facility is constructed in accordance with the UBC (ICBO 1994) and DOE Order 6340 1A, with egress requirements conforming to NFPA 101, Life Safety Code. Facility layout is discussed in Chapter B2.0.

The structural systems typically were constructed from one or more of the following materials: 3,000 lb/in reinforced concrete with reinforcing steel and welded wire fabric, carbon steel structural steel (ASTM A36/A36M-97a), structural tubing (ASTM A500-96 Grade B), steel pipe (ASTM A53-97 Type E or S Grade B), welding material (AWS D1 1-96, E70XX electrode), common bolts (ASTM A307-94), high-strength bolts (ASTM A325-97), nuts (ASTM A563-96), and hardened washers (ASTM F436-93). All masonry and concrete designs comply with the current UBC (ICBO 1994) ACI-207, Mass Concrete, ACI-318, Building Code Requirements for Reinforced Concrete, ACI-224, Control of Cracking, ACI-349 Code Requirements for Nuclear Safety Related Concrete Structures, and ACI-350 Environmental Engineering Concrete Structures, where applicable, and Precast Concrete Institute standard MNL-17 Manual of Quality Control for Plants and Production of Architectural Precast Concrete Products.

The CVDF building was designed to adequately resist the effects of various load combinations (dead load, live load, and accident load or seismic and wind load). The governing documents utilized in determining load combinations for performance category 3 structures were:


The following load combinations were used to determine the demand on reinforced concrete members of the CVDF building

1. $D_e = 14D + 17L + 17H$
2. $D_e = D + L + H + T + E$
3. $D_e = 0.9\ [D + L + H + T + W_t]$
4. $D_e = D + L + H + E$
5. $D_e = D + 13W$
6. $D_e = 105D + 13L + 13H + 105T$
7. $D_e = 14D + 17L + 17H + 17W$
8. $D_e = 105D + 13L + 13H + 13W$

where

$D_e$ = demand
$D$ = dead load including collateral dead load where it produces the more critical condition
$L$ = live load including applicable crane impact loads snow load
$T$ = normal operational thermal load
$W$ = wind load
$W_t$ = wind load due to tornado
$E$ = earthquake load
$H$ = earth pressure load

Where any load reduced the effects of other loads, the load factor for that load was taken as zero unless that load can be demonstrated as always present and in such a case the load factor was taken as 0.9.

The following load combinations were used to determine the demand on structural steel members of the CVDF building

<table>
<thead>
<tr>
<th>Load Combination</th>
<th>Stress Limit Coefficient$^b$</th>
</tr>
</thead>
<tbody>
<tr>
<td>1 $D_e = D + L + L_T + T$</td>
<td>$1.0^c$</td>
</tr>
<tr>
<td>2 $D_e = D + L + L_T + T + E$</td>
<td>$1.6^d$</td>
</tr>
</tbody>
</table>

$^b$Stress limit coefficients were applied to primary stress limits given in ANSI/AISC N690 Sections Q1 5 1, Q1 5 2, Q1 5 3, Q1 5 4, Q1 5 5, Q1 6, Q1 10 and Q1 11
$^c$For primary plus secondary stress the allowable limits were increased by a factor of 1.5
$^d$The stress limit coefficient in shear was not allowed to exceed 1.4 in members and bolts
where all symbols have the same meanings as above except

\[ H = \text{soil pressure, which shall be combined with live load in the combinations above} \]
\[ L = \text{live loads, including applicable crane impact loads but excluding snow or roof live loads} \]
\[ L_T = \text{larger of roof live load or snow load} \]

All load combinations were checked for the no live-load condition

The loads used in the design of the structures are listed in the Design Requirements Document (HNF-SD-SNF-DRD-002) The following design assumptions were incorporated in the structural analyses performed

- Allowable basic soil bearing pressure is 71.8 kPa (1,500 lb/ft²) based on sandy soils (soil classification SW, SP, SM, SC, GM and/or GC) in accordance with the UBC (ICBO 1994) Allowable soil bearing increases for depth and width of footing are applied for a maximum allowable pressure of 215.5 kPa (4,500 lb/ft²)
- Coefficient of friction between soil and foundation is 0.25
- Active soil pressure for unrestrained conditions is 561 kg/m²/m (35 lb/ft²/ft)
- Active soil pressure for restrained conditions is 882 kg/m²/m (55 lb/ft²/ft)
- Modulus of elasticity of soil is 1,380 kPa (200 lb/in²)

Static structural calculations were performed on the building to ensure structural loads (live loads and dead loads) along with severe wind loads (straight winds and tornado winds) were accounted for in the design and analysis. The tornado wind loads provided the largest load

*Secondary stresses that were used to limit primary stresses were treated as primary stresses. In no instance was the allowable stress permitted to exceed 0.70 \( F_u \) (in axial tension, 0.7 \( F_u \) times the ratio \( Z/S \) for tension plus bending \( F_u \) is the specified minimum tensile strength of the material \( Z \) is the plastic section modulus, and \( S \) is the elastic section modulus)
demands on the structure. Two special situations were identified and evaluated to ensure that the effects of tornado wind loads were properly accounted for:

1. When the tornado wind load is applied to the front of the process bay structure, the interior walls of the process bays will flex more than the exterior walls and cause an increase in the loads on the exterior walls. A calculation was performed to show that the walls can withstand this additional loading from the effect of differential rigidity between the walls.

2. The adjacent administration area building and transfer corridor wing are not designed to withstand tornado wind loads and potential interaction with the process bays was evaluated.

Structural calculations demonstrating adequacy of the facility design are included in SNF-3001.

Metal roof decks are stitch-welded with a membrane exterior. Roofing complies with the UBC Chapter 15 (ICBO 1994) for roof construction and covering, Chapter 16 (ICBO 1994), for roof and wind design, and Factory Mutual Class I or UL Class A for materials. For the administration area and process water tank rooms, Factory Mutual Data Sheet 1-28S and the Factory Mutual Approval Guide were used for acceptable materials selection and construction practices. The minimum roof slope is 0.25 in/ft.

Each process bay includes an overhead bridge crane with a capacity of 4,000 lb. The crane is designated performance category 1 and is designed in accordance with the criteria included in CMAA 74 Specifications for Top Running and Under Running Single Girder Electric Overhead Traveling Cranes Utilizing Under Running Trolley Hoist, with class D heavy service rating. The bridge trolley and hoist are driven by electric motors and are capable of operation from the ground floor or mezzanine level.

Differential pressure indication is provided in conjunction with the reference air system and is monitored and controlled by the automatic temperature control system. Additional safety-significant differential pressure switches with alarms to the control room are also provided. This will initiate operator action if process or HVAC changes must be initiated based on confinement status.

B4 4.1.3 Functional Requirements The following functional requirements are identified in the accident analyses described in Chapter B3.0.

Natural Phenomena The CVDF process bay structure must provide performance category 3 resistance to natural phenomena hazard events to protect the process equipment and to accomplish 3-over-1 seismic qualification to preclude damage or impairment of safety-class systems or components. Of specific concern for 3-over-1 protection are the process hood support stand and its flexible tubing connections to the process connectors attached to the MCO, the tempered water supply piping to the cask, the SCHe system, and the SCIC system. The process...
water tank room must provide performance category 2 resistance to natural phenomena hazards. The cask trailer must be positioned properly in the process bay based on calculational results for sway of the top of the MCO to protect process line connections and the process support stand.

Confinement To accomplish the confinement function, the passive presence of the physical structures must assist in mitigating a radioactive release within the CVDF after a DBA by allowing the ventilation systems to maintain a negative pressure in the process bays and process water tank room (as applicable) during facility operations. A specific leak rate for the structure is not critical to the passive confinement function required. The physical structure provides a space for the ventilation systems to exhaust. The ventilation system maintains a negative pressure within the process bay structures and the process water tank room, which is the safety parameter to be ensured. The structural confinement function is required while the process general supply/exhaust and process bay local exhaust HVAC systems are operating.

Bridge Crane The process bay crane shall provide 3-over-1 seismic protection and shall not be allowed to move over the cask-MCO during processing activities.

B4.1.4 System Evaluation The following evaluation of the process general supply/exhaust HVAC system demonstrates that the functional requirements for that system have been accomplished by the system design.

Natural Phenomena The process bays are designed and constructed as performance category 3 structures. The internal structural components (e.g., the mezzanines in the process bays) also are constructed as performance category 3 structures. The process water tank room is designed and constructed as a performance category 3 structure but is now designated performance category 2. The process support area is also designed and constructed as a performance category 2 structure, including the second floor mechanical equipment room. The administrative area is designed and constructed as a performance category 1 structure. There are distinct structural boundaries between performance category 2 and performance category 3 structural interfaces. Performance category 3 SSCs are isolated from the effects of performance category 2 SSC failures, such that the performance category 3 safety functions can still be accomplished under design basis conditions. The administrative building is structurally decoupled from the process bay structure and designed to fail away from the process bay area during a performance category 3 event. The control room is not needed for response after natural phenomena events.

The design basis natural phenomena loads and conditions applied to the structural design of the performance category 3 structures in the CVDF are tabulated in Table 1 of WHC-SD-SNF-DB-010 and listed in Table B1-1. These criteria are applied to SSCs based upon the performance category 2 or performance category 3 classification. Straight wind load and missile load criteria are given in WHC-SD-SNF-DB-010 Chapter 3. Tornado missiles have been eliminated from the CVDF requirements. DBE loads and spectra are given in WHC-SD-SNF-DB-010 Section 2.2. The design requirements for ashfall are given in WHC-SD-SNF-DB-010, Section 2.2. The CVDF does not have performance category 4 SSCs.
The design basis river flood level assuming a 25% breach of Grand Coulee Dam is 460 ft, compared with the 474-ft elevation of the CVDF site; therefore, flooding will not occur. The CVDF structures are designed to resist earth and groundwater loads in accordance with DOE Order 6430 1A and DOE-STD-1020-94 as administered using the guidance of HNF-PRO-097. Dynamic earth pressures were considered in the design of performance category 3 below-grade structures.

The CVDF secondary enclosure (process bays, performance category 3 and process water tank room, although now designated performance category 2) were structurally evaluated to seismic criteria for performance category 3 structures, with analyses documented in Calculation 96004 12 (Revision 1), Seismic Evaluation of Cold Vacuum Drying Facility Secondary Enclosure and Process Water Tank Room (Advent, 1997). These dynamic analyses used three-dimensional finite element modeling (ANSYS [SASI 1989]) to evaluate combined dead, live, and seismic loads. The CVDF seismic analysis modeled and analyzed the facility using the general purpose finite element program. The model is three-dimensional and consists of beam, shell, and mass elements as appropriate. Fixed-base static analyses results were used to address dead and live loads. A 20 lb/ft² roof live load mass was included in the seismic analysis to simulate the mass from the snow load.

In accordance with the recommendations of DOE-STD-1020-94 for performance category 3 facilities, equivalent static analyses may be performed if a single degree-of-freedom representation of the structure can be justified or if it can be shown that the fundamental frequency in each direction is greater than the cut-off frequency of the corresponding response spectrum. As these criteria cannot be met for the CVDF process bays nor for the process water tank room, a dynamic analysis method was used in the evaluation of the structures. Though a transient dynamic analysis is permitted, a response spectrum analysis approach is recommended by DOE-STD-1020-94 and was used here. The response spectrum technique requires modal analysis of the structure to determine its natural frequencies and the corresponding mode shapes. The dynamic characteristics of the structure are used in calculating structural responses such as displacements, forces and moments and stresses caused by externally applied seismic forces as defined by the response spectra.

HVAC duct systems and other mechanical and electrical systems attached to the CVDF structures have been addressed in the seismic analysis by including their mass in the dead load inertia. By definition the rigid equipment items that are a part of these systems do not require a coupled analysis with the primary structure. The flexible (i.e., less than 33 Hz) components of these systems (e.g., HVAC ducting) contribute less than 1% to the mass of the primary structure, and consequently, according to the guidance offered in ASCE-4-86, Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety-Related Nuclear Structures, coupling between these secondary systems and the primary structure model is not necessary.

A fixed base seismic analysis was performed because the building foundation is shallow and the response spectra include the effects of free-field soil amplification and are peak-broadened to
account for soil–structure interaction-induced frequency shifts. A fixed base analysis neglects the benefits of foundation scattering (kinematic interaction) and radiation damping. The potentially detrimental effects of rocking modes induced by inertial soil–structure interaction also are excluded in this approach; however, these effects are insignificant for low buildings. The peaks of the response spectra have been broadened in the high frequency range (conservatively) to address the potential for frequency shifting associated with soil–structure interaction. This peak-broadening of the response spectra was performed in response to the requirements of DOE-STD-1020-94, which states that it is permissible to ignore the beneficial effects of soil–structure interaction and assume that the DBE ground motion applies at the foundation level of the structure, provided that any frequency shifting (primarily of the fundamental frequency of the structure) caused by soil–structure interaction is considered. The results of a multimodal three-dimensional seismic analysis do not readily allow for the identification of a single "fundamental frequency," and the extent of frequency shifting caused by soil–structure interaction is not known, so a conservative broadening of the response spectra was performed for all frequencies above that corresponding to the peak.

In accordance with the recommendations of DOE-STD-1020-94, the response spectra correspond to response level 3 damping because such damping is expected to be reached before structural failure. The response level 3 damping value recommended for welded and friction-bolted metal structures is 7%. The response level 3 damping value recommended for prestressed concrete structures is also 7%. Consequently, the response spectra (Figure B1-11) used are taken from WHC-SD-SNF-DB-010 for 7% damping.

The confinement requirements of maintaining a differential pressure do not apply for the performance category 3 DBE event, therefore, there is no limit to the amount of acceptable cracking. DOE-STD-1020-94 states that when evaluating the structural adequacy of the SSC, response level 3 damping may be used in elastic response analyses independent of the state of response actually reached because such damping is expected to be reached before structural failure. In addition, $F_p$, the inelastic energy absorption factor, is conservatively taken as one

Seismic response from modes with frequencies larger than one-half the zero period acceleration frequency, $f_a$, is calculated in accordance with ASCE-4-86. For each direction of seismic excitation, significant (mode coefficient ratio equal to or greater than 0.01) structural modes with frequencies less than the 16 5 Hz threshold frequency (flexible modes) are combined using the grouping method of NRC Regulatory Guide 1.92, Combining Modal Responses and Spatial Components in Seismic Response Analysis, as recommended in ASCE-4-86. Flexible modes are divided into groups in which the highest frequency of any mode in the group is less than or equal to 1.1 times the lowest frequency of any mode in the group. Groups are formed beginning with the first mode of the structure and ascending through the remaining flexible modes in order of frequency. Each flexible mode is contained in only one group. Responses from modes that fall into a particular group are summed absolutely to obtain the response of that group. Responses from groups are combined by taking the square root of the sum of the squares.
DOE-STD-1020-94 requires that the demand due to accidental horizontal eccentricity be included in the total seismic response. This requirement is included to account for the possible added effect of inclined seismic waves. The accidental eccentricities are specified as equal to 5% of the building dimension in each of the two horizontal directions. The response from accidental eccentricity is combined absolutely with the modal and spatial combinations response.

Response of the CVDF finite element model to seismic loading identifies drift and demand/capacity results as discussed in the following paragraphs. DOE-STD-1020-94 specifies an allowable story drift of 0.010 times the story's height when both shear and flexure contribute to drift and 0.004 times the story's height when shear deflection is the primary contributor. The maximum lateral displacement of any node in the model is 0.431 inches, well below 0.72 inches (0.004 x 180 inches [height to mezzanine level]).

In accordance with DOE-STD-1020-94, demand/capacity evaluation for reinforced concrete components of CVDF performance category 3 structures are evaluated according to the specifications of ACI 349-90 and structural steel components are evaluated according to the specifications of ANSI/AISC N690. For the precast, prestressed, reinforced concrete wall panels, final design was accomplished by the panel vendor. The seismic analysis provided load requirements on the panels, panel-to-panel connections, panel-to-footing connections and attachments to the wall that must be sustained for the D + L ± S (where D is dead, L is live, and S is seismic) load combinations. For the panels themselves, the required level of prestress in the vertical direction was determined and the horizontal axial load and moment about a vertical axis that must be sustained were provided.

The steel beams and columns in the process cell area (secondary enclosure) and in the process water storage tank room were evaluated against the requirements of ANSI/AISC N690. The member categories are tabulated on Table B4-9. For each member category (real constant number), the results were reviewed for the two seismic load combinations (D + L + S, D + L - S) and the tornado wind load D + W, W, (where W, is tornado wind load). The maximum absolute value of each load combination was determined for each member category. Table B4-9 contains a summary of the demand/capacity ratios determined for combined axial and bending stresses in each member category.

The precast prestressed, concrete wall panels that enclose the process area and the process water storage tank room were evaluated against the requirements of ACI 349-90. Each of the walls is tabulated on Table B4-10. For each identified wall, the results of the two seismic load combinations (D + L + S and D + L - S) were reviewed. The maximum value of each load combination was determined for the elements composing each wall. Each wall panel was evaluated to determine the level of prestress required to ensure that in the prestressed direction (vertically) there would be no tensile stresses developed under the subject load combinations. A level of prestress was determined for each wall and provided in the seismic report (Advent 1996). Each panel also was evaluated to determine the combination of horizontal direct load and bending moment about the vertical axis that must be sustained by the horizontal rebar provided by the panel vendor. The horizontal loads and moments about a vertical axis are also provided in the seismic report for each wall.
<table>
<thead>
<tr>
<th>Element</th>
<th>Component</th>
<th>Demand/capacity ratio</th>
<th>Governing condition</th>
</tr>
</thead>
<tbody>
<tr>
<td>Steel members axial plus bending stress</td>
<td>CM W12X65</td>
<td>0.34</td>
<td>D+L S</td>
</tr>
<tr>
<td></td>
<td>CM W12X87</td>
<td>0.93</td>
<td>D+Wt</td>
</tr>
<tr>
<td></td>
<td>BM W14X48</td>
<td>0.92</td>
<td>D+L S</td>
</tr>
<tr>
<td></td>
<td>BM W12X40</td>
<td>0.83</td>
<td>D+Wt</td>
</tr>
<tr>
<td></td>
<td>BM W8X10</td>
<td>0.88</td>
<td>D+L S</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1.07</td>
<td>D+L+S</td>
</tr>
<tr>
<td></td>
<td>BM W12X19</td>
<td>0.96</td>
<td>D+L+S</td>
</tr>
<tr>
<td></td>
<td>BM W10X17</td>
<td>0.82</td>
<td>Wt</td>
</tr>
<tr>
<td></td>
<td>BM W16X40</td>
<td>0.77</td>
<td>D+L</td>
</tr>
<tr>
<td></td>
<td>BM W12X50</td>
<td>0.97</td>
<td>D+Wt</td>
</tr>
<tr>
<td></td>
<td>CM TSX5X1/4</td>
<td>0.30</td>
<td>D+L</td>
</tr>
<tr>
<td></td>
<td>CM TS4X4X1/4</td>
<td>1.29 (0.585)</td>
<td>D+L S</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.51</td>
<td>D+L+S</td>
</tr>
<tr>
<td></td>
<td>BM W16X77</td>
<td>0.96</td>
<td>D+Wt</td>
</tr>
<tr>
<td></td>
<td>BR W12X26</td>
<td>0.91</td>
<td>D+L+S</td>
</tr>
<tr>
<td></td>
<td>BR W12X35</td>
<td>0.97</td>
<td>D+Wt</td>
</tr>
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<td></td>
<td>BR W27X94</td>
<td>0.50</td>
<td>D+Wt</td>
</tr>
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<td></td>
<td>BR L4X4X1/4</td>
<td>0.68</td>
<td>D+Wt</td>
</tr>
<tr>
<td></td>
<td>BR L2X2X3/16</td>
<td>0.84</td>
<td>D+L S</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1.02</td>
<td>D+L+S</td>
</tr>
<tr>
<td></td>
<td>BMR W8X21</td>
<td>0.67</td>
<td>D+Wt</td>
</tr>
<tr>
<td></td>
<td>BMR W14X22</td>
<td>0.92</td>
<td>D+Wt</td>
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<td></td>
<td>CMR W8X24</td>
<td>0.37</td>
<td>D+L S</td>
</tr>
<tr>
<td></td>
<td>BMR C12X25</td>
<td>0.87</td>
<td>D+Wt</td>
</tr>
<tr>
<td></td>
<td>BMR C8X11 5</td>
<td>0.83</td>
<td>D+Wt</td>
</tr>
<tr>
<td></td>
<td>BMR L2X2X3/16</td>
<td>0.85</td>
<td>D+L S</td>
</tr>
<tr>
<td></td>
<td>BMR L3X3X1/4</td>
<td>0.82</td>
<td>Wt</td>
</tr>
<tr>
<td></td>
<td>BMR C10X15 3</td>
<td>0.81</td>
<td>D+Wt</td>
</tr>
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<td>BMR C15X33 S</td>
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<td>Steel members shear</td>
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<td>CM W12X87</td>
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<td>BM W14X48</td>
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<td>BM W12X40</td>
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<td>BM W8X10</td>
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<td>BM W12X19</td>
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Table B4-9 Steel Beams and Columns Results (2 sheets)

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<tr>
<th>Element</th>
<th>Component</th>
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<th>Governing condition</th>
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<td>BM W16X40</td>
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<td>BM W12X50</td>
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<td></td>
<td>CM TS5X5X1/4</td>
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<td>D+L±S</td>
</tr>
<tr>
<td></td>
<td>CM TS4X4X1/4</td>
<td>0.01</td>
<td>D+L±S</td>
</tr>
<tr>
<td></td>
<td>BM W16X77</td>
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<td>D+L±S</td>
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<td>BR W12X35</td>
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<td>D+L±S</td>
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<td></td>
<td>BR W27X94</td>
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<td>BR L4X4X1/4</td>
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<td>D+L±S</td>
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<tr>
<td></td>
<td>BR L2X2X3/16</td>
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<td>D+L±S</td>
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<td>D+L±S</td>
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<td>BMR W14X22</td>
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<td>BMR C8X11 5</td>
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<td></td>
<td>BMR L2X2X3/1</td>
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<td>BMR L3X3X1/4</td>
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<td>D+L±S</td>
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<td>BMR C10X15 3</td>
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<td>D+L±S</td>
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</table>


*The ratio was computed based on maximum component forces of each group. The value of 1.07 was determined based on envelope loading. For actual member loading, value is determined to be less than 1.0.

Four members attached to wall line E were detached to reduce demand load.

*TS4X4X1/4 members were replaced with TS5X5X1/4 supporting mezzanine floor at elevation 34.9 m (114 ft 6 in) (new demand/capacity ratio was hand calculated as 0.585).

*Due to conservatism in ZPA, 2% over is acceptable in secondary members.

D = dead load
CM = column in the processing area
BM = beam on the mezzanine floor of the processing area
BR = beam on the roof of the processing area
BMR = beam in the water tank area
CMR = column in the water tank area
L = live load
S = seismic seismic demand from design basis earthquake
Wt = wind tornado demand
ZPA = zero period acceleration
## Table B4-10  Prestressed Panels Evaluation (3 sheets)

**Concrete wall force and moment summary**

<table>
<thead>
<tr>
<th>Wall line</th>
<th>fx (lb/in)</th>
<th>fy (lb/in)</th>
<th>mx (in lb/in)</th>
<th>my (in lb/in)</th>
<th>tsx (lb/in)</th>
<th>tsy (lb/in)</th>
<th>ts/V</th>
</tr>
</thead>
<tbody>
<tr>
<td>Wall line A</td>
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<td></td>
<td></td>
<td></td>
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<tr>
<td>D+L S minimum</td>
<td>1398 60</td>
<td>382 43</td>
<td>2 774 30</td>
<td>1 478 40</td>
<td>195 76</td>
<td>166 39</td>
<td>0 6790651</td>
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<tr>
<td>D+L+S maximum</td>
<td>671 26</td>
<td>337 26</td>
<td>2 791 40</td>
<td>1 451 80</td>
<td>195 91</td>
<td>183 30</td>
<td></td>
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<td>Design load</td>
<td></td>
<td></td>
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<td></td>
</tr>
<tr>
<td>Vertical preload</td>
<td></td>
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<td></td>
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<tr>
<td>Horizontal load</td>
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<td></td>
<td></td>
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<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>F = ±400 lb/in M = 1 500 in lb/in</td>
<td></td>
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<tr>
<td>D+L S minimum</td>
<td>837 09</td>
<td>505 21</td>
<td>2 443 10</td>
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<tr>
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<td>423 65</td>
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<td>1 390 40</td>
<td>139 09</td>
<td>46 02</td>
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<tr>
<td>Design load</td>
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<tr>
<td>Horizontal load</td>
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</tr>
<tr>
<td>F = ±600 lb/in M = 1 400 in lb/in</td>
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<tr>
<td>D+L S minimum</td>
<td>361 73</td>
<td>1 001 90</td>
<td>1 993 80</td>
<td>3 290 00</td>
<td>140 23</td>
<td>226 07</td>
<td>0 783606</td>
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<td>552 15</td>
<td>1 975 90</td>
<td>3 213 30</td>
<td>149 00</td>
<td>215 23</td>
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<td>Horizontal load</td>
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</tr>
<tr>
<td>F = ±842 15 lb/in</td>
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<td>Wall line 7</td>
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<tr>
<td>D+L S minimum</td>
<td>1 004 70</td>
<td>965 21</td>
<td>2 670 60</td>
<td>1 194 50</td>
<td>248 86</td>
<td>55 70</td>
<td>0 9171591</td>
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<tr>
<td>D+L+S maximum</td>
<td>737 73</td>
<td>1 074 70</td>
<td>2 744 30</td>
<td>1 428 10</td>
<td>264 60</td>
<td>58 99</td>
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<td>Design load</td>
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<tr>
<td>Vertical preload</td>
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<td></td>
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<tr>
<td>Horizontal load</td>
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<tr>
<td>F = ±1 100 lb/in M = 1 500 in lb/in</td>
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### Table B4-10  Prestressed Panels Evaluation  (3 sheets)

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<thead>
<tr>
<th>Wall line E</th>
<th>Concrete wall force and moment summary</th>
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<tr>
<td></td>
<td>fx (lb/in)</td>
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<td></td>
<td>Horizontal load</td>
</tr>
<tr>
<td></td>
<td>Out-of plane shear</td>
</tr>
</tbody>
</table>

| Wall line C7 line 1 and line F in monorail | Concrete wall force and moment summary | |
|-------------------------------------------|---------------------------------------|-
| D+L S minimum | 1217 00 | 727 68 | 373 98 | 1027 90 | 54 65 | 35 30 | 0 189411 |
| D+L+S maximum | 1064 20 | 439 93 | 400 12 | 1117 10 | 44 82 | 52 50 |
| Design load | Vertical preload | 1557 03 lb/in |
|              | Horizontal load | F = ± 1300 lb/in, M = 500 in lb/in |

| Wall line 2 in monorail | Concrete wall force and moment summary | |
|-------------------------|---------------------------------------|-
<p>| D+L S minimum | 751 94 | 959 77 | 722 86 | 1434 30 | 75 51 | 796 97 | 27624652 |
| D+L+S maximum | 946 46 | 525 36 | 970 36 | 1229 30 | 60 23 | 198 15 |
| Design load | Vertical preload | 1959 66 lb/in |
|              | Horizontal load | F = ± 1000 lb/in, M = 1000 in lb/in |
|              | Out-of plane shear | FV = ± 800 lb/in |</p>
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<tr>
<th>Wall line</th>
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<th>Maximum</th>
<th>Minimum</th>
<th>Maximum</th>
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<td>3991.29</td>
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<td>16835.67</td>
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<td>26350.64</td>
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<td>Wall line 3 4 5 6</td>
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<td>6949534</td>
<td>3832.79</td>
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<td>Wall line 7</td>
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<td>7098911</td>
<td>21390</td>
<td>2607311</td>
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<td>Wall line E</td>
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<td>22570.5</td>
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<td>Wall line C 7 2 F</td>
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<td>17978.5</td>
<td>151081</td>
<td>27452.59</td>
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<td>Wall line 1</td>
<td>11468</td>
<td>12468.3</td>
<td>18312.6</td>
<td>432978</td>
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<table>
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<tr>
<th>f_x</th>
<th>f_y</th>
<th>f_z</th>
<th>f_x</th>
<th>f_y</th>
<th>f_z</th>
</tr>
</thead>
</table>

f_x = axial force along the member x-axis (lb/ft)
f_y = axial force along the member y-axis (lb/ft)
m_x = moment causing bending stress in the element x-direction (lb/ft)
m_y = moment causing bending stress in the element y-direction (lb/ft)
s_x = out-of-plane shear force on surface with x-direction normal (lb/ft)
s_y = out-of-plane shear force on surface with y-direction normal (lb/ft)
s = shear force
V = shear demand
D = dead load
L = live load
S = seismic load
F = force
M = momentum
Each wall panel also was evaluated for out-of-plane shear. An effective depth of 0.8 x 6 in = 4.8 in was used in the shear evaluation in accordance with the recommendations of ACI 349-90. The shear stresses were evaluated against a capacity limit of $\phi f'_c$, where $\phi = 0.85$ (in accordance with Section 9.3 of ACI 349-90 and $f'_c$ is 34,475 kPa (5,000 lb/in$^2$). Such a capacity limit (half the concrete's capacity in shear) forgoes the requirement for minimum shear reinforcement. The resulting demand/capacity ratios for out-of-plane shear on the wall panels are reported in Table B4-10. Five wall panel elements require out-of-plane shear reinforcement. They are the central 10-ft-wide panels in each of the bays between column lines 2 and 3, 3 and 4, 4 and 5, and 5 and 6 for the wall along column line E and the western-most panel of the wall along column line 2 (at column line F). The line E panels require out-of-plane shear reinforcement from an elevation of 109 ft, 0 in to 119 ft, 0 in, for a distance of 7 ft, 0 in from its northmost edge. For the panel along column line 2, the entire panel must be reinforced for the defined out-of-plane shear. The required level of out-of-plane shear that must be sustained by these panels is reported in Table B4-10.

The wall panel connection loads between the wall panels and the footing for each connection point are reported in the Table B4-10 tabulation of loads. Units are in pounds-force.

Member connections between steel members and member connections between steel members and the prestressed concrete wall panels were also analyzed. The following maximum absolute reaction forces were determined for the two load combinations ($D + L + S$ and $D + L - S$) and listed in Table B4-11. The connections specified in the wall panel design were extensively analyzed and are adequate for performance under seismic loadings.

Footings supporting performance category 3 structures were evaluated for the two seismic load combinations ($D + L + S$ and $D + L - S$). Results are summarized in Table B4-12.

Column base reaction loads were reviewed and the maximum tensile and shear forces are reported in Table B4-13. By comparison with the uplift ($F_u$) loads reported in structural analyses (see SNF-3001 Calculation MEI-2288-ST), uplift is controlled by the wind load combination. Therefore, by inspection, the base plates evaluation for the seismic load combinations are not necessary.

Structural recommendations resulting from the analyses were incorporated into the building design, thus providing a structure that meets seismic design requirements for performance category 3.

Additional dynamic analyses (Advent 1998) were conducted to develop in-structure seismic response spectra as a basis for seismic analysis of attached performance category 3 systems and equipment. The process hood support structure was subsequently redesigned and analyzed separately as a performance category 3 structure, as reported in SNF-5038 Seismic Analysis of CVDF Process Hood Support Structure and Mounting Frame.
Table B4-11  Steel Member Connection Analysis Results

<table>
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<tr>
<th>Drawing number</th>
<th>Connection detail</th>
<th>Demand/capacity ratio</th>
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<td>H 1-82126 Section A</td>
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<td>Not reported</td>
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<td>H 1-82126 Section B</td>
<td>Weld</td>
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<td>D+L+S</td>
</tr>
<tr>
<td>H 1-82126 Section E</td>
<td>Weld</td>
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<td>D+L+S</td>
</tr>
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<td>H 1-82129 Section 4 5 6 7</td>
<td>Weld</td>
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<td>D+L+S</td>
</tr>
<tr>
<td>H 1-82130 Section A and B</td>
<td>Weld</td>
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<td>D+L+S</td>
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<td>H 1-82130 Section C</td>
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<td>D+L+S</td>
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<tr>
<td>H 1-82130 Section D and E</td>
<td>Bolted shear</td>
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<td>0.077</td>
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<tr>
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<td>Bolted - shear</td>
<td>0.27</td>
<td>D+L+S</td>
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<tr>
<td></td>
<td></td>
<td>0.09</td>
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<td>Welded</td>
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<tr>
<td>H 1-82130 Section D and E</td>
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D = dead load
L = live load
S = elastic seismic demand from design basis earthquake
### Table B4-12 Footing Results

<table>
<thead>
<tr>
<th>Footing number</th>
<th>Maximum bearing pressure (lb/in²)</th>
<th>Allowable soil bearing pressure (lb/in²)</th>
<th>Demand/capacity ratio</th>
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<tbody>
<tr>
<td></td>
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<td></td>
<td>Bending moment</td>
</tr>
<tr>
<td>FTG-2</td>
<td>21 37</td>
<td>38 79</td>
<td>0 626</td>
</tr>
<tr>
<td>FTG-3</td>
<td>29 59</td>
<td>30 48</td>
<td>0 895</td>
</tr>
<tr>
<td>FTG-4</td>
<td>15 64</td>
<td>33 25</td>
<td>0 527</td>
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<tr>
<td>FTG-5</td>
<td>15 17</td>
<td>38 79</td>
<td>0 554</td>
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<td>FTG-6</td>
<td>12 73</td>
<td>38 79</td>
<td>0 935</td>
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<tr>
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<td>14 12</td>
<td>33 25</td>
<td>0 667</td>
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<td>24 94</td>
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<td>FTG-15</td>
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<td>S-FTG(E)</td>
<td>21 84</td>
<td>22 17</td>
<td>0 213</td>
</tr>
<tr>
<td>S-FTG(INT)</td>
<td>20 39</td>
<td>19 40</td>
<td>0 086</td>
</tr>
</tbody>
</table>

**Notes**

1. The seismic calculations for the footings were conservatively performed assuming a soil bearing capacity of 1 500 lb/ft². Subsequent soil studies demonstrated that the soil at Cold Vacuum Drying Facility could bear 5 300 lb/ft² providing significant margin when compared to calculated results.

### Table B4-13 Maximum Column Base Tensile and Shear Force Summary

<table>
<thead>
<tr>
<th>Column</th>
<th>F&lt;sub&gt;x&lt;/sub&gt; (lb)</th>
<th>F&lt;sub&gt;y&lt;/sub&gt; (lb)</th>
<th>F&lt;sub&gt;z&lt;/sub&gt; (lb)</th>
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</thead>
<tbody>
<tr>
<td>W12x87</td>
<td>5,347</td>
<td>2,655 6</td>
<td>4,772 4</td>
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<td>W12x65</td>
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<td></td>
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<tr>
<td>TS4x4x¼</td>
<td>No uplift load</td>
<td></td>
<td></td>
</tr>
<tr>
<td>TS5x5x¼</td>
<td></td>
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<td></td>
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<tr>
<td>W8x24</td>
<td>147 6</td>
<td>2,621 8</td>
<td>296 0</td>
</tr>
</tbody>
</table>

F<sub>x</sub> = axial force along the member x-axis lb/in
F<sub>y</sub> = axial force along the member y axis lb/in
F<sub>z</sub> = axial force along the member z axis lb/in
Seismic evaluations of the transportation trailer system at the CVDF, with resulting effects on the CVDF floor slab, were conducted as documented in HNF-2179 Seismic Analysis of TN-DE&SH Cask Transportation Trailer System for the CVDF Facility. The cask transporter has been analyzed for sway on the rubber tires and suspension to demonstrate the safety-class process lines will not be damaged. The seismic rocking of the cask-MCO will not impose accelerations on the MCO greater than the analyzed transportation loads (HNF-SD-TP-SARP-017) because the suspension is active, the trailer is not anchored, and there are no impacts with other SSCs. No seismic tie-down of the trailer is required, but specific positioning requirements for the centerline of the cask-MCO are identified. The cask is to be centered on column row C of the building in the east-west direction and the centerline of the rollup door in the north-south direction. A tolerance of ± 4 in has been specified.

There is no defined operating basis earthquake for the CVDF and no small-magnitude near-field earthquake analysis has been performed. Performance category 2 earthquake loads are those given by the UBC (ICBO 1994) for seismic zone 2B for essential facilities (DOE-STD-1020-94). Engineering calculations demonstrating the adequacy of the performance category 2 CVDF structural design are listed in SNF-3001.

The site storm and flood water drainage system was designed for a performance category 3 event. This is in conformance with DOE Order 5480 28 and its supporting document DOE-STD-1020-94. As discussed in WHC-SD-SNF-DB-010, the CVDF is also designed and constructed to have a level of safety comparable to an NRC-licensed facility. This requirement resulted in additional flood criteria. Engineering calculations demonstrating the adequacy of the CVDF soils to drain the design basis rainfall are listed in SNF-3001.

Confinement. The passive presence of the physical structures is required so the HVAC system can accomplish a differential pressure within the facility. This is ensured by the general-service CVDF structures because they were designed and constructed to performance category 3 criteria to accomplish natural phenomena hazard mitigation and 3-over-1 seismic performance for the protection of safety-class systems and components. The telescoping doors must be closed to accomplish the confinement function while processing an MCO in a bay. Engineering calculations demonstrating the adequacy of the CVDF structural design to accomplish 3-over-1 protection of all safety-class and safety-significant SSCs are listed in SNF-3001.

The structures allow the ventilation systems to maintain a negative pressure in the process bays and process water tank room during processing operations. The physical structure provides a space for the ventilation systems to exhaust. The safety-significant function of maintaining a negative pressure within the process bay structures and process water tank room does not require the normal air balance differential pressure be maintained but simply requires a negative pressure. Building confinement is required during normal process operation. Inleakage much greater than normal rates could be tolerated and still accomplish this safety function requirement as long as a negative pressure is maintained by ventilation exhaust flow. Because the ventilation systems are not required to operate after a DBE the confinement function of the structures is not required after the DBE.
**Bridge Crane**  The process bay bridge cranes are classified as general service. The process bay cranes are also classified as NRC important-to-safety category C in accordance with HNF-SD-SNF-DB-003. Controls are implemented in accordance with general-service requirements and the cranes are designated performance category 1. Engineering calculations demonstrating the adequacy of the CVDF process bay cranes to accomplish performance category 1 criteria and 3-over-1 protection of all safety-class and safety-significant SSCs are listed in SNF-3001. TSR controls are implemented to preclude crane activity over the safety-class process lines and tempered water supply piping during processing activities to preclude common mode failures.

**B4 4 1 5 Controls (Technical Safety Requirements)**  The following assumptions associated with the CVDF process bay and process water tank room structures require TSRs to ensure performance of the safety function:

- The process bay telescoping door is closed prior to and during MCO processing in a bay.
- The trailer shall be positioned properly in the process bay (see Section B4 4 1 4) before connecting process systems.
- The process crane shall not be allowed to move over the cask-MCO during processing activities.

**B4 4 2 Process General Supply/Exhaust Heating, Ventilation, and Air Conditioning System**

**B4 4 2 1 Safety Function**  The process general supply/exhaust HVAC system has been identified in Chapter B3 0 to perform the following safety functions to mitigate safety-significant consequences from the bounding accident scenarios. Only the exhaust portion of the system is credited with mitigation (HEPA filtration) and maintenance of differential pressure to accomplish confinement:

- The process general exhaust system provides confinement of radioactive material within the process bays and process water tank room during processing operations if a release from primary confinement occurs. Its operation mitigates the radioactive release from the CVDF after the gaseous release, liquid release, MCO external hydrogen explosion, MCO internal hydrogen explosion, and MCO overpressurization accidents (identified in Table B4-8) by maintaining a negative pressure in the process bays and process water tank room during facility operations, as indicated in Table B4-8, and by HEPA filtering the building air prior to discharge from the facility.
The process general exhaust isolation dampers at each process bay must fail closed upon facility loss of power to enable the process bay local exhaust HVAC and process vent system operating on standby power to maintain differential pressure in the process bays. This functional requirement is applicable for the gaseous release, MCO external hydrogen explosion, MCO internal hydrogen explosion, and MCO overpressurization accidents.

B4 4 2 2 System Description

Although the supply portion of the general supply/exhaust HVAC system supplies conditioned air to the CVDF process bay and process support areas and plays a role accomplishing normal air balance, it does not perform a safety function (as identified in Chapter B3 0) and is therefore not addressed here. The exhaust portion of the system is credited with safety-significant mitigation (HEPA filtration) and maintenance of differential pressure to accomplish confinement.

Each process bay is served by the general exhaust system, with sufficient capacity to ensure an adequately controlled ventilation flow, as required to contain contamination in the process areas. Ventilation flows are shown in Figure B2-29.

Air from the transfer corridor, associated support rooms, the process water tank room, and the mechanical room is also exhausted into the general exhaust system. The system maintains exhaust air flow and a negative pressure differential in the process bays with respect to all other areas external to the bays. The automatic temperature control system regulates the supply air inlet damper of the process bay recirculation HVAC system to control differential pressure in each bay.

Isolation dampers are provided on all duct branches connecting to the general exhaust system to prevent backflow in the event of exhaust system shutdown. These dampers fail closed on loss of air or electricity and are interlocked to the fire alarm system to close upon any fire indication in each bay. The exhaust filter enclosure is a bag-in bag-out type, three filters wide by four filters high. The four rows of HEPA filters are required for normal operation. Each row can be isolated for filter changes. The air is filtered with an 85% American Society of Heating, Refrigerating, and Air-Conditioning Engineers (ASHRAE) prefilter and two stages of HEPA filters before discharging the exhaust air via the exhaust stack. The HEPA filters conform to ANSI/ASME N509, Nuclear Power Plant Air-Cleaning Units and Components. There are three test sections located within the filter box to accomplish testing in accordance with ANSI/ASME N510, Testing of Nuclear Air Cleaning System. Equipment and ductwork that make up the process general supply/exhaust HVAC systems are supported and anchored in accordance with the requirements necessary to meet the criteria of performance category 2 construction. Most ductwork for the general exhaust system is fabricated of round galvanized steel. A fire screen is located before the HEPA filter enclosure. Instrumentation in the duct includes humidity monitoring, with remote alarm and temperature/rate of rise alarm and interlock to shut down the exhaust fans. Redundant 30 horsepower variable speed exhaust fans are provided for the general exhaust system, but backup power is not provided. Flow indication ahead of the HEPA filters controls the exhaust fan speed, to compensate for filter loading by...
maintaining constant volume flow. The automatic temperature control system monitors all HVAC functions. The airflow through the general exhaust system is normally uncontaminated. Only during upset conditions (i.e., accidents) would radioactive contamination be handled by the general exhaust system. Such conditions include releases into a process bay or the process water tank room. Local readout and remote differential pressure alarms to the control room are provided from each process bay (bays 2, 3, 4, and 5) and from the process water tank room by the reference air system.

The specific components of the general exhaust ventilation system subject to designation as safety-significant components (based on the analysis in Chapter B3.0) are as follows:

- Pneumatic isolation dampers on the exhaust ducts from the process bays
- All ducting and fittings necessary to route airflow to the HEPA filters
- Exhaust HEPA filters and filter differential pressure indication (local)

**B4 4 2 3 Functional Requirements** The following functional requirements have been identified in the accident analyses described in Chapter B3.0

**Ductwork** The process general exhaust system ductwork system must route discharge air to the HEPA filters from any process bay conducting normal process operations and from the process water tank room any time the PWC pumps are operating.

**HEPA Filters** The process general exhaust system HEPA filter installation must accomplish a 99.9% filter efficiency, as credited in the Chapter B3.0 accident analyses, any time the process general exhaust system is operating.

**Differential Pressure** The general exhaust branch at each process bay or process water tank room conducting process operations, must accomplish confinement of radioactive material for the process bays and the process water tank room by maintaining a negative building pressure and by flowing discharge air through HEPA filters. Nominal setpoints are provided in Section B4 4 5 2.

**Bay Isolation** The process general exhaust isolation dampers at each process bay must fail closed upon facility loss of electrical power or loss of instrument air to enable the process bay local exhaust HVAC and process vent system operating on standby power to maintain differential pressure in the process bays and preclude back-flow and communication between process bays.

**B4 4 2 4 System Evaluation** The following evaluation of the process general supply/exhaust HVAC system demonstrates that the functional requirements for this system have been accomplished by the system design.

**Ductwork** Assurance that a filtered discharge path is available is accomplished by the performance category 2 qualification of the process general exhaust system. The performance category 2 general exhaust duct was designed in accordance with ERDA 76-21, *Nuclear Air*...
Cleaning Handbook schedule methodology The ductwork integrity is a credited design feature demonstrated by normal differential pressure in the building.

HEPA Filters A system filter decontamination factor of at least $1 \times 10^3$ is accomplished by the two installed stages of testable 99.97% efficient HEPA filtration prior to discharge to the CVDF stack. The filters are tested in accordance with ANSI/ASME N510 upon installation and annually thereafter.

Differential Pressure Confinement of radioactive material in the process bays (bay 2, 3, 4, and 5), and the process water tank room is assured by maintaining a negative building pressure in each area. This is demonstrated by the differential pressure instruments and alarms provided in the control room for each of these areas. Each process bay and the process water tank room is individually alarmed for differential pressure by the reference air system. The automatic temperature control system monitors HVAC functions and modulates the volume damper on the process bay recirculation HVAC system outside air inlet to maintain the designated differential pressure.

Bay Isolation The process general exhaust isolation dampers at each process bay fail-closed upon facility loss of electrical power or loss of instrument air to enable the process bay local exhaust HVAC and process vent system operating on standby power to maintain differential pressure in the process bays and to preclude back-flow and communication between the process bays. The nuclear grade dampers are in accordance with ANSI/ASME N509 Class 1 Construction Class B, to accomplish a “bubbletight” rating.

B4 4 2 5 Controls (Technical Safety Requirements) The following assumptions associated with the process general exhaust system require TSRs to ensure performance of the safety function:

- The HEPA filters are tested in accordance with ANSI/ASME N510 upon installation and annually thereafter.
- The isolation damper and fail-closed actuator for any process bay are functional during any processing operation in any process bay. The exhaust isolation dampers shall fail closed upon loss of electrical power and loss of air.

B4 4 3 Process Bay Local Exhaust Heating, Ventilation, and Air Conditioning and Process Vent System

B4 4 3 1 Safety Function The process bay local exhaust HVAC and process vent system is identified in Chapter B3 0 as performing the following safety functions to prevent or mitigate safety-significant consequences from the bounding accident scenarios.
Annex B — Cold Vacuum Drying Facility

- Provides confinement of radioactive material released within the process bay during processing operations. The system's operation mitigates the radioactive release from the CVDF after the gaseous release, MCO external hydrogen explosion, MCO internal hydrogen explosion, and MCO overpressurization accidents by directing the flow from the process bays through ductwork to the HEPA filters, prior to discharge from the facility. The hood isolation dampers fail closed and are provided with a instrument air reservoir for operation on standby power. The local exhaust system fans can restart with standby power and can maintain a process bay differential pressure when the process general supply/exhaust HVAC system is not operable (e.g., facility loss of electrical power).

- Maintains local exhaust flow sufficient to dilute potentially hydrogen-rich process gas discharges into the local exhaust duct to preclude flammable mixtures from being generated. Its operation prevents an external hydrogen explosion accident by diluting hydrogen gas from the process vent lines and directing the flow from the process bays through ductwork to HEPA filters prior to discharge from the facility for external and internal hydrogen explosion, thermal runaway, and MCO overpressurization accidents. The 30 lb/in² gauge vent line discharge as a result of the MCO overpressurization accident is also adequately diluted by the local exhaust flow. The hood isolation dampers fail closed on loss of power. The local exhaust system can restart with standby power and can reestablish local exhaust flow within 1 minute sufficient to dilute potentially hydrogen-rich process gas discharges into an exhaust duct and maintain a process bay differential pressure when the process general supply/exhaust HVAC system is not operable (i.e., facility loss of electrical power). The local exhaust provides a process hood low-flow alarm to the control room to preclude cask venting with inadequate dilution flow. The local exhaust also provides the cask venting connection with a flow-restricting orifice along with a shut-off valve interlocked to a local exhaust low-flow switch for each bay.

Table B4-8 summarizes the confinement functions performed by the local exhaust HVAC and process vent system in conjunction with the CVDF structures, to mitigate or prevent the DBAs.

**B4 4 3 2 System Description** The process bay local exhaust system is provided to serve four process bays. This system serves the MCO process hood and the process vent streams from each process bay and the PWC receiver tank. The process vents discharge directly into the local exhaust duct. Ventilation flows are shown on Figure B2-29. These vents are associated with the VPS cask venting system, tempered water (annulus) system tank vents, the SCHe discharges, and the 30 lb/in² vent. Hydrogen-rich streams utilize double-check valves to preclude ductwork air from entering the vent pipes. The process vents are open to the HEPA filters via the system ductwork during operation and when the system shuts down. The process vent connections are downstream from the isolation dampers provided on each duct branch connecting to the process hood. These isolation dampers fail closed upon loss of electrical power, and each damper actuator is provided a safety-significant, 8 3-gal, compressed-air reservoir to operate the damper.
during standby power operation. The isolation dampers are interlocked to close upon fire alarm signal in each process bay.

Redundant safety-significant exhaust fans of the process bay local exhaust system are provided standby power, which will reestablish flow within 1 minute. Each fan has 100% capacity to operate the local exhaust system. Safety-significant backflow dampers downstream from the exhaust fans preclude backflow into the fan that is not operating. The automatic temperature control system monitors all normal HVAC functions. Local exhaust system operation on standby power does not require control signals because all functions operate directly from the standby power system and restart circuit.

The ductwork system contains a flow switch that provides a low-flow alarm in the control room for each process bay duct branch which provides operators with information about system functionality.

All ductwork for the process bay local exhaust system is fabricated of round stainless steel duct or pipe and is of welded construction. The exhaust filter box is a bag-in bag-out type. Two filters wide by three filters high. The air is filtered with an 85% ASHRAE prefilter and two stages of HEPA filters, before discharging the exhaust air via the CVDF exhaust stack. The HEPA filters are in conformance with ANSI/ASME N509 with three test sections located within the filter box to accomplish testing in accordance with ANSI/ASME N510. All three rows of HEPA filters are required for normal operation. Each row can be isolated for filter changes. A fire screen is located before the HEPA filter enclosure. Instrumentation in the duct includes humidity monitoring with remote alarm and temperature/rate of rise alarm and interlock to shut down the exhaust fans. Redundant 15 horsepower variable speed exhaust fans are provided. Flow indication ahead of the HEPA filters controls the exhaust fan speed to compensate for filter loading, by maintaining constant volume flow. Equipment and ductwork that make up the process bay local exhaust HVAC system are supported and anchored in accordance with the requirements necessary to meet the criteria of performance category 2 construction.

The process bay local exhaust HVAC and process vent system provides special connections and dilution flow for cask and MCO venting. The process bay local exhaust system is required to provide this function during initial cask venting, using the removable cask vent jumper connected from the cask vent port to the process vent cask venting quick-disconnect fitting or for special-case MCO venting prior to processing using the MCO venting tool attached to the cask venting connection. The process bay local exhaust system is designed to provide sufficient airflow in conjunction with the flow restriction orifice and the flow valve interlocked to the low flow alarm in the cask venting connection to the ductwork in each bay, to dilute the maximum potential hydrogen release from the cask to less than the lower flammability limit for hydrogen in air.
The specific components of the process bay local exhaust HVAC and process vent system subject to designation as safety-significant components (based on the analysis in Chapter B3 0) are as follows:

- Local exhaust fans and plenum
- All ducting and fittings necessary to route airflow to the filters
- Exhaust HEPA filters and filter differential pressure (local)
- Instrumentation (flow switches and alarms) necessary to verify adequate airflow are functional
- Isolation dampers in each bay that fail closed
- Process vent lines for cask venting including orifice and flow valve interlocked to the low flow alarm in each bay
- Process vent lines for the 30 lb/in² vent path, SCHe, and VPS discharges

B4 4 3 3 Functional Requirements

**Fan operation** The process bay local exhaust system must be operating and accomplish adequate dilution of hydrogen introduced during the cask venting and processing activities to ensure that the hydrogen lower flammability limit is not reached.

**Flow alarm** A flow switch in each process bay branch to the process hood shall be provided with a remote alarm to the control room. It must be demonstrated that the flow in any process bay is adequate prior to initiating the cask venting operation, and during MCO processing in each process bay.

**HEPA filters** The process bay local exhaust system HEPA filter installation must accomplish a 99.9% filter efficiency as credited in the Chapter B3 0 accident analyses.

HEPA filter loading shall be administratively controlled to less than 9.4 g of spent fuel while in service. The HEPA filter enclosure shall have the capability to accurately take repeatable measurements for the monitored radiation dose from the prefilters and HEPA filters. Measurements may be performed with portable instruments. Physical access to minimally shielded surveillance locations must be provided.

**Ductwork** The process bay local exhaust system ductwork must route discharge air to the HEPA filters from any process bay conducting normal process operations. The process bay local exhaust system must provide a discharge path via the ductwork and HEPA filter for the SCHe pressure venting and the 30 lb/in² vent path.
**Hood isolation damper** Each isolation damper located near the process hood must fail closed upon loss of electrical power. The dampers shall have the ability to re-open while the general-service instrument air is not operating.

**Cask venting** A flow restriction orifice and a flow valve interlocked to a low-flow switch in the cask venting connection is incorporated into the ductwork to preclude flammable mixtures in the ductwork and vent lines.

**Standby operation** The process bay local exhaust HVAC and process vent system in the restart mode must be capable of maintaining differential pressure within the process bays under all design basis HEPA filter loading conditions without instrumentation or controls. It must also meet the minimum flow requirement of 1,000 ft³/min for dilution.

**B4 4 3 4 System Evaluation** The following evaluation of the process bay local exhaust HVAC and process vent system demonstrates that the functional requirements for this system have been accomplished by the system design.

**Fan operation** The process bay local exhaust HVAC and process vent system has a high degree of reliability. Redundant safety-significant exhaust fans are provided for the process bay local exhaust system. Each fan has 100% capacity of the local exhaust system flow to accomplish confinement and dilution flow requirements. The process bay local exhaust HVAC and process vent system has a “hand-off-auto” station located at each exhaust fan. When the fan’s hand-off-auto station is in the ‘auto’ position, the automatic temperature control system monitors and controls the exhaust system. The “on” position is used for fan testing only and the switch is placed in the ‘auto’ position for normal operations. The exhaust fans each have a manual isolation damper that must be placed in the open position for proper operation of the system. Engineering calculations demonstrating the adequacy of the local exhaust fan capacities are compiled in SNF-3001. Backflow dampers downstream of the exhaust fans preclude backflow into the fan that is not operating. The local exhaust fans are provided standby power that will restart the system within 1 minute. Local exhaust system operation on standby power does not require control signals, as all functions operate directly from the standby power system and restart circuit. No accidents are identified in Chapter B3.0 that require performance category 3 seismic qualification of the exhaust fans for the process bay local exhaust HVAC and process vent system.

**Flow alarm** Adequate flow in any process bay during MCO processing and prior to initiating the cask venting operation is demonstrated using safety-significant flow switches in each branch. The flow switches provide a low-flow alarm in the control room for each process bay duct branch to inform operators of system functionality. The low flow switches activate at 1,150 ft³/min (above 1,120 ft³/min to account for a 10% error band of the flow switches per SNF-4451). A minimum flow of 1,000 ft³/min is required for dilution of process vent discharge per calculation SNF-4301.

**HEPA filters** A system filter decontamination factor of at least $1 \times 10^3$ is accomplished by either of the two installed stages of HEPA filtration, prior to discharge to the CVDF stack.
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time the process bay local exhaust system is operating. The air is filtered with an 85% ASHRAE prefilter and two stages of HEPA filters before discharging the exhaust air via the CVDF exhaust stack. The HEPA filters conform to ANSI/ASME N509, with three test sections located within the filter box to accomplish testing in accordance with ANSI/ASME N510. Differential pressure across each stage of HEPA filters is monitored and alarmed in the control room. Flow indication ahead of the HEPA filters controls the exhaust fan speed to compensate for filter loading by maintaining constant volume flow. Differential pressure instrumentation is provided to demonstrate HEPA filter loading and presence.

The HEPA filter loading is administratively controlled to less than 94 g of spent fuel while in service. The HEPA filter enclosure has the capability to accurately take repeatable measurements for the monitored radiation dose from the prefilters and HEPA filters. Measurements are performed with portable instruments within their calibration period. Physical access to minimally shielded surveillance locations is provided. This is accomplished by not exceeding a contact dose reading in the designated locations on the HEPA filter box of 82 mR/h (SNF-2770 Chapter 60), which incorporates a factor of 10 reduction in allowable filter loading to account for segregation on decay of gamma emitters (e.g., 82 mR/hr correlates to 9.4 g SNF).

A total of six dose readings shall be taken using an ion chamber instrument at designated points on the surface of the local exhaust filter box. These points are defined as the centerline of each of the three vertical filter rows for both the prefilter bank and the first HEPA filter bank. The readings shall be taken on the west side of the filter box at the center of each cover plate for that row of filters. A reading below 82 mR/h shall be observed at each survey point. These survey points shall be marked and numbered on the filter box for accuracy and repeatability. This accomplishes the dose reading credited in SNF-2770, Chapter 70 which states, “The dose points for the three filters were 2 in. to the side of the HEPA filter, corresponding to a likely dose rate measurement point for weekly surveys by radiation protection technicians.”

Ductwork All ductwork for the process bay local exhaust system is fabricated of round stainless steel duct or pipe and is of welded flange construction. Assurance that a discharge path is available to the HEPA filters is accomplished by the performance category 2 qualification of the process bay local exhaust system ductwork. The performance category 2 local exhaust duct was designed in accordance with ERDA 76-21 schedule methodology. Performance category 2 qualification is sufficient for discharge of the SCHe and 30 lb/in² vent path during performance category 3 events because pinching off these process vent lines in order to retain high pressures is not probable. Process bay confinement is not required after a seismic event.

Hood isolation damper The hood isolation dampers are pneumatically operated nuclear-grade dampers. The nuclear-grade dampers are in accordance with ANSI/ASME N509 Class 1 Construction Class B, to accomplish a “bubbletight rating”. Each process hood isolation damper fails closed upon loss of electrical power. The damper actuators are provided a safety-significant, compressed-air reservoir for operation on standby power. The isolation dampers are interlocked to close upon fire alarm signal in each process bay.
Cask venting Dilution of hydrogen introduced during the cask venting activity to ensure that the hydrogen lower flammability limit is not reached is accomplished by the system normal operating capacity of 1,300 ft³/min in conjunction with the cask venting orifice. A minimum flow of 1,000 ft³/min is required for dilution during cask venting. This is interlocked to the flow valve in the cask vent line to interrupt venting on loss of flow. The flow rate requirement and orifice diameter are documented in SNF-4301. For a 150 lb/in² gauge cask venting activity, an orifice of 0.041 in results in a hydrogen concentration of 1% in the local exhaust duct.

Standby operation The process bay local exhaust HVAC and process vent system in the restart mode with standby power is capable of maintaining differential pressure within the process bays under all design basis HEPA filter loading conditions without instrumentation or controls. The motor starter is energized by the standby power system and bypasses the variable speed drive for the fan, operating it at maximum capacity which is adequate for all HEPA filter loadings. Each process hood isolation damper re-opens using the safety-significant, compressed-air source provided by the compressed-air reservoir in each process bay to restart on standby power. The control signal is provided by the standby power restart circuit. To accomplish sufficient differential pressure, the isolation dampers on the general exhaust system and the outside air inlet to the process bay recirculation HVAC system must fail closed. These dampers and their safety-significant function are addressed in their appropriate system sections. HVAC flow rate and differential pressure evaluations for the standby operation of the local exhaust system are documented in SNF-3001, Calculation ME1 2621 ME 6.

B4 4 3 5 Controls (Technical Safety Requirements) The following assumptions associated with the process bay local exhaust system require TSRs to ensure performance of the safety function:

- The system minimum flow requirement of 1,000 ft³/min is met while processing an MCO in that process bay.
- The cask venting orifice is properly installed in the system.
- The installed flow switch and remote low-flow alarm on the process bay branch of the ductwork is functional in each process bay during MCO processing in that bay.
- The flow switch interlock to cask venting valve is functional.
- The HEPA filters are tested in accordance with ANSI/ASME N510 upon installation and annually thereafter.
- The HEPA filter inventory is less than 9.4 g of spent fuel to protect the assumption in the Chapter B3 0 accident analysis. (This is accomplished by not exceeding a contact dose reading in the designated locations on the HEPA filter box of 82 mR/h as described in Section B4 4 3 4.)
The isolation damper fail-closed actuator for any process bay is functional during any processing operation in that process bay.

The system restart capability with standby power is functional to reestablish minimum flow within 1 minute.

### B4 4 4 Process Bay Recirculation Heating, Ventilation, and Air Conditioning System

#### B4 4 4.1 Safety Function

The process bay recirculation HVAC system is analyzed in Chapter B3 0 to perform the following safety function to mitigate safety-significant consequences from the bounding accident scenarios:

- The system supply isolation dampers at each process bay must fail closed upon facility loss of power to enable the process bay local exhaust HVAC and process vent system operating on standby power to maintain differential pressure in the process bays. This is applicable for the gaseous release, MCO internal hydrogen explosion, MCO external hydrogen explosion, and MCO overpressurization accidents.

#### B4 4 4.2 System Description

The only portions of the process bay recirculation HVAC system that performs a safety-significant function are the fail-closed inlet isolation dampers. The isolation dampers are pneumatically operated nuclear-grade dampers located inside process bays 2 through 5 at the 24-in by 18-in supply air duct through the west wall of each bay. The external elevation is just above the roof of the mechanical equipment room. The pneumatic actuators fail to the closed position upon loss of electrical signal or loss of instrument air.

#### B4 4 4.3 Functional Requirements

**Supply isolation damper** Each inlet isolation damper located at the process bay west wall must fail closed upon loss of electrical power or loss of instrument air to support restart of the local exhaust system on standby power and establish differential pressure in the process bay.

#### B4 4 4.4 System Evaluation

**Supply isolation damper** Each supply inlet isolation damper fails closed upon loss of electricity or loss of instrument air. This supports restart of the local exhaust on standby power to establish differential pressure in the process bay because no HVAC control functions are provided by standby power, and the differential pressure calculations for standby power operation were performed with these dampers closed. HVAC flow rate and differential pressure evaluations for the standby operation of the local exhaust system are documented in SNF-3001 (Calculation MEI 2621 ME 6). The inlet isolation dampers are pneumatically operated nuclear-grade dampers. The nuclear-grade dampers are in accordance with ANSI/ASME N509, Class 1 Construction Class B, to accomplish a “bubbletight” rating.
**B4 4 4 5 Controls (Technical Safety Requirements)** The following assumptions associated with the process bay recirculation HVAC system requires TSRs to ensure performance of the safety function

- The supply inlet isolation damper and fail-closed actuator for any process bay is functional during any processing operation in any process bay

**B4 4 5 Reference Air System**

**B4 4 5 1 Safety Function** The reference air system is analyzed in Chapter B3 0 to perform the following safety functions to mitigate safety-significant consequences from the bounding accident scenarios

- The process bay and process water tank room differential pressure alarms provide a safety-significant function for mitigation of the consequence of gaseous release, liquid release MCO external hydrogen explosion MCO internal hydrogen explosion, and MCO overpressurization accidents. The alarm notifies personnel of the loss-of-confinement function, and personnel initiate appropriate action to preclude releases during the abnormal confinement condition. This requirement to demonstrate HVAC confinement function related to location and DBA is summarized in Table B4-8

- The system must provide a reliable reference air signal to the process bay and process water tank room differential pressure indicators with low differential pressure alarms so they can perform their safety-significant function

**B4 4 5 2 System Description** The process bays and the process water tank room are maintained at a slight negative pressure with respect to atmosphere and the remainder of the CVDF by the process general supply/exhaust HVAC system. The reference air system provides the capability to continuously monitor the static pressure relative to atmospheric pressure, of various spaces throughout the facility. The differential pressure-indicating transmitters (DPITs) indicate whether the HVAC confinement systems are functioning properly. The automatic temperature control system provides visual indication and alarms (general service) for differential pressure conditions in the control room. Safety-significant DPITs with local indication and remote indication of low-differential pressure alarm set points are also provided for the process bays and the process water tank room.

The reference air system consists of a series of static DPITs located throughout the CVDF that are interconnected by 0.5-in, 1-in, or 2-in copper tubing. Five manual valves along the reference air header are provided for maintenance and isolation purposes. The pressure differential is monitored in the control room and locally at each DPIT in the process water tank room, the transfer corridor and access rooms, the process bays, the mechanical equipment room, and the administration access corridor. The automatic temperature control system monitors all...
reference air pressures  Separate safety-significant DPITs and alarm circuits are also provided for
the process bays and the process water tank room as part of the reference air system.

The nominal set points are as follow:

- Process bays, -0 20 in water gauge
- Process water tank room, -0 25 in water gauge
- Transfer corridor access rooms, -0 15 in water gauge
- Transfer corridor and mechanical equipment rooms, -0 10 in water gauge
- Administration access corridor, +0 01 in water gauge

Differential pressure indication for HVAC control is considered general service and is
provided in conjunction with the reference air system and monitored by the automatic temperature
control system. Safety-significant DPITs with low differential pressure alarms to the control
room are also provided. The reference air header is also safety-significant to provide a reliable
reference air signal to the differential pressure alarms. This will initiate operator action if process
or HVAC changes must be initiated based on confinement status.

**B4 4 5 3 Functional Requirements** The following functional requirements have been identified
to ensure the reference air system performs its safety-significant function.

**Differential pressure instruments** DPITs for each process bay (bays 2, 3, 4, and 5) and
the process water tank room shall be provided with low differential pressure alarms in the control
room.

**Differential pressure alarms** The low differential pressure alarms must be able to
remotely indicate to the control room a loss of negative building pressure relative to atmospheric
pressure in the process bays (2, 3, 4, and 5), during MCO processing, and in the process water
tank room, while the PWC recirculation pump is operating.

**Reference air header** The reference air header and static pressure sensor must be
available and functional to provide a reliable reference air signal whenever the DPITs and low
differential pressure alarms are required to be functional.

**B4 4 5 4 System Evaluation** The reference air system provides DPITs alarms, and the
reference air header for safety-significant low differential pressure alarms. Each of the system
functional requirements is discussed below.

**Differential pressure instruments** Safety-significant DPITs with low differential pressure
alarm set points are provided in the process bays and the process water tank room. These DPITs
are in addition and redundant to the DPITs provided for the general-service HVAC control via
the automated temperature control system.
Differential pressure alarms  Safety-significant differential pressure alarms are provided to the control room to allow timely response to ventilation upset conditions. Engineering calculations demonstrating the adequacy of the performance category 2 reference air low differential pressure alarm system design are listed in SNF-3001.

Reference air header  The reference pressure is measured by a static pressure sensor located above the administration building roof. The reference air pipe header is fabricated of noncorrosive material (copper). Piping and instruments that make up the reference air system are supported in accordance with the requirements necessary to meet the criteria of performance category 2 construction. Engineering calculations demonstrating the structural adequacy of the performance category 2 reference air piping and installation design are listed in SNF-3001.

B4 4 5 5  Controls (Technical Safety Requirements)  The following assumptions associated with the reference air system require TSRs to ensure performance of the safety function:

- The system must be functional during processing activities in any process bay and whenever the process water pumps are operating.
- The DPITs and alarms must be able to detect a loss of negative building pressure relative to atmospheric pressure in the process bays (bays 2, 3, 4, and 5) and the process water tank room during processing activities in any of those areas.
- The process bay ventilation systems are maintaining normal differential pressures prior to initiating MCO processing in a bay.
- The process water tank room ventilation system is maintaining normal differential pressure prior to initiating MCO draining operations and at all times that any PWC pumps are operating.

B4 4 6  Instrument Air System

B4 4 6 1  Safety Function  The instrument air system is analyzed in Chapter B3.0 to perform the following safety functions to mitigate safety-significant consequences from the bounding accident scenarios:

- Provide a safety-significant function of being capable of re-opening the fail-closed process hood isolation dampers when the general supply/exhaust HVAC system is not operable and the hood exhaust fans are re-started under standby power. The system’s operation supports mitigation of potential radioactive releases from the CVDF after the gaseous release, MCO external hydrogen explosion, internal hydrogen explosion, and MCO overpressurization accidents. This requirement to demonstrate HVAC confinement functions for the process bays and each DBA is summarized in Table B4-8.
**B4 4 6 2 System Description**  The instrument air system is used to provide and distribute quality air throughout the CVDF Instrument air is supplied by duplex two-stage piston type compressors, a 250-gal receiver tank pressure control valve, instrument air dryer filters, and desiccant instrument air dryers. For normal operation, the instrument air system provides instrument air at a pressure of 100 lb/in² gauge to the HVAC isolation dampers.

HVAC system isolation dampers fail closed if there is loss of electrical power. To ensure the process hood dampers on the process bay local HVAC exhaust and process vent system are reopened when standby power is activated and the local exhaust fan restarts, safety-significant instrument air is provided to each air line to each process bay hood isolation damper. The safety-significant portion of the instrument air system in each process bay consists of an air tank, double-check valves in the inlet air supply line to each instrument air tank, piping from the tank to the hood isolation damper actuator, and a pressure-indicating gauge on the tank. Each instrument air tank reservoir has minimum capacity of 8.3 gal and is designed for a maximum pressure of 200 lb/in² gauge. The normal operating pressure of 100 lb/in² gauge (per vendor information) provides sufficient air volume in the reservoir to complete four actuations of the isolation damper. Double-check valves prevent loss of air to possible line breaks in the general service portions of the instrument air system and local air pressure indication is provided for the operator to monitor the tank pressure in each bay. The safety-significant reservoir tanks were constructed to ASME Section VIII (ASME 1995), and the safety-significant piping and valves were fabricated to ASME B31.3 and B31.9, *General Services Piping Code*.

On loss of power, the solenoid valve will block the flow of air to the actuator for the process hood isolation damper, and the damper will close. When standby power is activated, the solenoid valve is energized and will reopen, and air from the in-line safety-significant instrument air reservoir will flow to the actuator to open the damper as required. The general-service instrument air system normally maintains pressure in the reservoirs. The reservoirs are isolated from the instrument air system by safety-significant check valves when process upsets occur, which could result in loss of instrument air supply or loss of power. Periodic monitoring of the pressure in the tanks by the operators ensures the capability exists for functional operation of the dampers upon loss of power.

**B4 4 6 3 Functional Requirements**  The following functional requirements have been identified to ensure the specific portions of the instrument air system perform their safety-significant functions.

- **Operating Capacity** — Provide the safety-significant source of instrument air to provide the capability of opening the process hood fail-closed isolation dampers when standby power is activated. Have sufficient instrument air capacity to operate the process hood isolation damper actuator for four actuation cycles and the capability for operator verification of the tank air pressure.
System Operability, Calibration, and Testing — Ensure the instrument air system is operable, the reservoir is pressurized at all times, the reservoir tank pressure gauge is reading correctly, and there is periodic testing to demonstrate system functionality.

Seismic Qualification — Provide an instrument air supply for each process hood damper that is seismically qualified to performance category 2 criteria.

B4 4 6 4 System Evaluation Portions of the instrument air system provide an independent air supply for each process hood isolation damper. Each of the system functional requirements is discussed below.

Operating Capacity Under normal operation the instrument air system in each bay will be supplied at a pressure of 100 lb/in² gauge. Design of the instrument air systems for the hood isolation dampers is based on the air tanks having minimum capacity of 8 3 gal at a pressure of 100 lb/in² gauge to support isolation damper operation in a loss of power situation. Double-check valves are installed on the inlet air supply line to the instrument air tanks to provide system isolation from general-service portions of the system. This 8 3 gal air supply at 100 lb/in² gauge will provide a sufficient amount of air for four actuations (only one actuation required) of the process hood isolation damper actuators (per vendor information) that require 80 lb/in² gauge to open. Gauges on each of the air supply tanks provide a means for the operators to monitor the tank pressure and verify sufficient pressure is available for damper actuator operation.

System Operability, Calibration, and Testing Operation of each instrument air system is ensured through periodic testing of the standby power system and the process hood isolation dampers. Surveillance and in-service inspections of the instrument air system safety-significant components are conducted in accordance with operating procedures for respective components. The normal instrument air pressure is 100 lb/in² gauge, and the damper actuator requires 80 lb/in² gauge to operate. A periodic pressure hold test after closing the in-line instrument air valve to the reservoir demonstrates adequate leak tightness of the system. A pressure drop of less than 2 lb/in² gauge during a 4-hour hold test (0 5 lb/in² gauge/h 12 lb/in² gauge total drop over 24 hours) demonstrates that the reservoir can provide sufficient air supply for 24 hours to hold the local exhaust damper open. Additional surveillance of reservoir pressure during standby power operation ensures continued function. The calibration of the tank pressure gauges is conducted according to the manufacturer’s recommendations.

Seismic Qualification Piping and instruments that make up the safety-significant portion of the instrument air system are constructed and supported in accordance with the requirements necessary to meet the performance category 2 criteria. Engineering calculations demonstrating the structural adequacy of the performance category 2 instrument air reservoir and piping system design are listed in SNF-3001.
B4 4 6 5 Controls (Technical Safety Requirements) The following assumptions associated with the instrument air system require TSRs to ensure performance of the safety function

- The instrument air system in each bay must be functional whenever the process bay local HVAC exhaust and process vent system is operating in that bay

- Leak tightness of the instrument air system piping, tank and check valve components (pressure hold test showing less than 0.5 lb/in² gauge per hour drop) is verified on an appropriate schedule to ensure operation if there is loss of instrument air supply

- Instrument air system tank gauges are calibrated on an appropriate schedule

- Operators periodically verify tank pressure is above 90 lb/in² gauge pressure by observing the reservoir pressure gauges according to procedure

B4 4 7 Standby Power System

B4 4 7 1 Safety Function The standby power system provides the safety-significant function of supplying AC electrical power to the process bay local exhaust HVAC and process vent system to provide confinement in the process bays and dilution flow within the local exhaust duct during facility loss of electrical power incidents. The standby power system has been analyzed in Chapter B30 to perform the following safety functions to mitigate safety-significant consequences from the bounding accident scenarios

- Provide standby power to maintain process bay local exhaust HVAC and process vent system flow and/or to accomplish sufficient differential pressure in the process bays for confinement during an electrical outage that occurs during or after a gaseous release, external or internal hydrogen explosion, thermal runaway reaction, or MCO overpressurization accidents

- Provide a restart circuit to perform the functions necessary to restart the process bay local exhaust HVAC and process vent system, including the fan motor logic and process hood isolation damper control during or after the gaseous release, MCO external or internal hydrogen explosion, thermal runaway reaction, or MCO overpressurization accidents

B4 4 7 2 System Description The 100 kW diesel generator system supplies 3-phase power at 480 V standby power to selected equipment through an automatic transfer switch. The automatic transfer switch selects the normal power source or the emergency generator for supplied loads based on the normal source voltage and frequency. An engine start signal is provided by the automatic transfer switch if the normal source voltage drops below a set level for a duration in excess of the set delay. A load bank is also provided to test the generator.
Generator startup is accomplished using an air start system that cranks the diesel engine to sufficient revolutions per minute to start the engine. The air start system uses its own general service air compressor to supply a 400-gal safety-significant air reservoir through a pressure control valve and double-check valves. A gauge is available on the line to the air reservoir to verify reservoir pressure.

Upon receiving the signal to start, the diesel generator will come up to the rated voltage and frequency. Once these parameters are obtained, the diesel generator will then supply the required power through an automatic transfer switch to the following selected loads:

- Process bay local exhaust HVAC and process vent system exhaust fan (safety significant) (only one fan runs at a time)
- Process hood isolation damper actuator solenoid valves in each bay (safety significant)
- Bypass transformer to pickup UPS loads if batteries are depleted
- Instrument air compressors (general service)
- Heat trace for process bay fire protection and deionized water lines (general service)

The heated diesel generator building consists of two rooms that house two diesel generators (only one generator is operational) an air start system, and the control panel for the operational diesel generator. The generator is fueled from a 50-gal day tank within the building. There is also a 500-gal tank supply integrated to automatically refill the day tank that will contain at least 150 gal of fuel to allow at least 24 hours of operation without refueling. The tank is located outside the generator building aboveground on a concrete pad. The tank is double walled with leak detection within the annulus space.

**B4 473 Functional Requirements** The following functional requirements have been identified to ensure the standby power system performs its safety-significant function:

**Diesel Generator** Provide sufficient standby power to supply designated safety-significant loads for a minimum period of 24 hours.

**Diesel Generator Building** The building shall be maintained at a temperature above 40 °F when the diesel generator is not operating to meet vendor recommendations for engine start. The diesel fuel in the day tank shall be maintained above 40 °F to support engine start capability.

**Restart Circuit** Provide a restart circuit to perform the functions necessary to restart the process bay local exhaust HVAC and process vent system, including fan motor logic and damper positioning, within 1 minute after loss of normal power. Also, provide power to the facility UPS.
loads air compressor motor, and process bay heat trace, which enhances the reliability of these general-service systems, which are not mandatory but provide defense in depth during standby power operation

System Operability, Calibration, and Testing Ensure the standby power system is operable, the system instrumentation is reading correctly, and periodic testing is conducted

Seismic Qualification Provide a safety-significant standby power system that is qualified to performance category 2 criteria for natural phenomenon

B4 4 7 4 System Evaluation The standby power system provides safety-significant functional requirements as discussed below

Diesel Generator The diesel generator has been procured to have the capability to provide 100 kW of standby power to be used for selected loads. An automatic transfer switch monitors the incoming power to the building and switches to standby power if there is a loss of normal electrical power. The engine and generator are rated for performance category 2 natural phenomena criteria. The 50-gal day tank and the 500-gal fuel storage tank provide sufficient fuel capacity for 24-hour operation. These parameters are documented in SNF-3001 Calculation MEI 2621 ME 7

Diesel Generator Building The building is insulated and maintained at a temperature above 50 °F by an electric heater with thermostatic control, to meet vendor recommendations for diesel start reliability. The diesel fuel day tank is inside the diesel generator building and is maintained at building temperature. This ensures the fuel and generator are at a temperature sufficient for reliable starting in accordance with vendor recommendations.

Restart Circuit The signal to start the backup power system comes from the automatic transfer switch that monitors normal power to the CVDF. The transfer to emergency power will be initiated upon reduction of normal power to 70% of nominal voltage. When the diesel engine starts and the generator voltage and frequency have obtained setpoint values, the automatic transfer switch will disconnect the generator from the load bank and transfer to the standby power loads. Transfer back to normal power shall occur when the normal electrical power source restores to 90% of nominal voltage.

The diesel engine controls are operated by a 12V DC lead acid battery that is continuously charged and mounted on the engine skid. The control panel has readouts for all pertinent engine and generator operating parameters. Startup of the engine is accomplished by use of an air start system that will initiate upon loss of normal power. The air start system relies on air from a 400-gal reservoir tank that is continuously kept at pressure by a standby power general-service air compressor. The air compressor supplies air through a regulator and double-check valves, and a gauge is provided to monitor reservoir pressure. Air starting will crank the engine until the speed exceeds 650 revolutions per minute or until 8 seconds elapses. The start sequence is repeated up to 6 times (reservoir pressure provides a minimum of 6 start attempts), within a delay period of

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less than 4 seconds. When the engine starts, the generator voltage will build as speed increases, until the voltage and frequency are within selected pick-up values (85% to 100% voltage, 90% to 100% frequency).

The selected loads to be supplied by standby power are to either of the local exhaust fans (only one operating at a time), the solenoid valves that control the actuators to the local exhaust hood isolation dampers, the facility UPS bypass transformer, and the instrument air compressors and the process bay heat trace. The local exhaust hood isolation dampers fail closed on initial loss of power and will move to the open position once power is restored to the solenoid valves, because the actuators are provided a safety-significant source of instrument air for the power loss situation. Restoration of power to the instrument air compressors provides a defense-in-depth capability to provide air for operation of the actuators. Restoration of power via the bypass transformer to the facility UPS loads provides defense-in-depth such that differential pressure instrumentation may be available during standby power operation although it is not required as calculations demonstrate sufficient differential pressure is maintained under the controlled conditions of standby power operation (SNF-3001, Calculation MEI 2621 ME 6).

**System Operability, Calibration, and Testing** Operation of the standby power system is ensured through periodic testing according to the manufacturer's recommendations. A load bank is provided with the standby power system to conduct engine startup and operational testing. Surveillance and in-service inspections of the standby power system safety-significant components are conducted in accordance with the manufacturer's recommendations for respective components and TSR requirements. Calibration of the instrumentation including the generator starter and reservoir pressure gauge is conducted according to the manufacturer's procedures and standards.

**Seismic Qualification** The engine generator piping, fuel storage tanks, and instruments that make up the standby power system were purchased and installed in accordance with the requirements necessary to meet the criteria of performance category 2 construction. Engineering calculations demonstrating the structural adequacy of the performance category 2 design and installation are listed in SNF-3001.

**B4 4 7 5 Controls (Technical Safety Requirements)** The following assumptions associated with the standby power system require TSRs to ensure performance of the safety-significant function:

- The standby power system must be capable of startup and operation at any time MCO processing is initiated in any process bay.
- Functional testing of the diesel generator and local exhaust fan startup using the standby power system has been performed on an appropriate schedule consistent with the respective manufacturer's schedule and practice.
The gauge on the diesel engine air starter reservoir is monitored on an appropriate schedule to ensure capability for standby power activation.

Maintenance is periodically performed on the diesel generator system per the manufacturer's recommendations.

The diesel generator fuel supply is sufficient for 24-hour operation and is verified on an appropriate schedule.

The diesel generator building is to be maintained above the minimum diesel starting temperature of 40°F when the diesel generator is not operating.

The diesel fuel temperature in the day tank and associated fuel lines are to be maintained above 40°F.

B4.4.8 Cold Vacuum Drying Facility Safety—Significant Special Tools

B4.4.8.1 Safety Function Two special tools are identified in Chapter B3 0 as performing the safety functions to mitigate safety-significant consequences from the bounding accident scenarios and for worker safety. These are the “cask vent jumper tool” and the “MCO vent jumper tool.” The safety functions of these tools are listed below.

- The cask vent jumper is to contain the flow of hydrogen from the cask headspace during venting such that the hydrogen is discharged into the process vent connection of the process bay local exhaust HVAC and process vent system and not discharged unrestricted into the process bay to preclude an external hydrogen explosion and protect the facility worker.

- The MCO vent jumper is to contain the flow of hydrogen from the MCO headspace during venting under abnormal conditions such that the hydrogen is discharged into the process vent connection of the process bay local exhaust HVAC and process vent system and not discharged unrestricted into the process bay to preclude an external hydrogen explosion and protect the facility worker.

B4.4.8.2 System Description The cask vent jumper is a flexible line with attached fittings to connect between the cask lid fitting and the local exhaust cask vent connection. Before removing the cask lid, the cask vent jumper is connected between the cask lid vent port and the local exhaust system cask vent connection. The cask connector allows a bolted gasketed connection to the cask lid with the ability to open the vent port valve after the connection is completed. The line confines any hydrogen concentrations that could be flammable or explosive, if released and mixed with air in an uncontrolled fashion.
The MCO vent jumper is a flexible line with attached process connector fittings and a 30 lb/in² gauge rupture disk that connects between the MCO filtered process exit port connection and the local exhaust cask vent connection. The 30 lb/in² rupture disk vents to the room. If there is an abnormal condition that requires venting of the MCO prior to making normal process connections, this flexible hose jumper can be attached to the MCO and pressure can be vented to provide extended time prior to initiating normal processing. The line confines any hydrogen concentrations that could be flammable or explosive if released and mixed with air in an uncontrolled fashion.

The safety-significant vent jumper tools are designed and fabricated in accordance with codes and standards identified in Table B4-3 for process equipment and valves.

B4 4 8 3 Functional Requirements

Hydrogen confinement The functional requirement for the cask vent jumper is to perform its safety function of confinement of hydrogen-rich gas without leakage or rupture at system operating pressures up to 150 lb/in² gauge

The functional requirement for the MCO vent jumper is to perform its safety function of confinement of hydrogen-rich gas, without leakage or rupture, at system operating pressures up to 150 lb/in² gauge.

B4 4 8 4 System Evaluation

Hydrogen confinement The cask vent jumper has been designed to meet its functional requirements and perform its safety functions. The flex hose is rated for 150 lb/in² gauge service. The flex hose interfaces with the process bay local exhaust HVAC and process vent system through the cask vent connection, which provides sufficient dilution flow rate in the exhaust duct to preclude an external hydrogen explosion.

The MCO vent jumper has been designed to meet its functional requirements and perform its safety functions. The flex hose is rated for 150 lb/in² gauge service. The rupture disk is rated at 30 lb/in². The rupture disk interfaces with the process bay local exhaust HVAC and process vent system through the cask vent connection, which provides sufficient dilution flow rate in the exhaust duct to preclude an external hydrogen explosion.

B4 4 8 5 Controls (Technical Safety Requirements) The following assumptions associated with the vent jumpers requires TSRs to ensure performance of their safety functions:

- The cask vent jumper is not damaged prior to cask venting in a bay
- The MCO vent jumper is not damaged prior to MCO venting in a bay
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Annex B — Cold Vacuum Drying Facility

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Figure B4-1  Cold Vacuum Drying Process Connections and Instrument Basics

Note  All Instruments PT, FIT TSH and LSL have two components per bay  
PT1 Pressure Transmitter (PT 1*36 1*37)  
PT2 Pressure Transmitter (Vacuum) (PT 1*08, 1*10)  
FIT Flow indicating Transmitter (FIT 1*20 1*21)  
TSH Temperature Switch High (TSH 1*28, 1*29)  
LSL Level Switch Low (LSL 1*24 1*28)  
TSHH Bay Temperature Switch High High (TSHH 1*38 1*39)  

Note The represents 2 through 8 for bays 2 8
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CHAPTER B5 0

DERIVATION OF TECHNICAL SAFETY REQUIREMENTS
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<th>Acronym</th>
<th>Description</th>
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<tr>
<td>AC</td>
<td>Administrative Control</td>
</tr>
<tr>
<td>BED</td>
<td>building emergency director</td>
</tr>
<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
</tr>
<tr>
<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>HEPA</td>
<td>high-efficiency particulate air (filter)</td>
</tr>
<tr>
<td>HVAC</td>
<td>heating, ventilation, and air conditioning</td>
</tr>
<tr>
<td>LCO</td>
<td>Limiting Condition for Operation</td>
</tr>
<tr>
<td>LCS</td>
<td>Limiting Control Setting</td>
</tr>
<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
</tr>
<tr>
<td>PWC</td>
<td>process water conditioning</td>
</tr>
<tr>
<td>SSC</td>
<td>structure, system, and component</td>
</tr>
<tr>
<td>TSR</td>
<td>technical safety requirement</td>
</tr>
<tr>
<td>VPS</td>
<td>vacuum purge system</td>
</tr>
<tr>
<td>mR</td>
<td>mili-roentgen</td>
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</table>
B5 0 DERIVATION OF TECHNICAL SAFETY REQUIREMENTS

B5 1 INTRODUCTION

A description of the essential features of Spent Nuclear Fuel (SNF) Project derivation of technical safety requirements (TSRs) is provided in Chapter 5 0 of the SNF Project Final Safety Analysis Report (FSAR)

B5 2 REQUIREMENTS

The requirements that form the basis for the SNF Project derivation of TSRs are identified in Section 5 2 of the SNF Project FSAR

B5 3 TECHNICAL SAFETY REQUIREMENTS COVERAGE

The group of TSRs for analyzed hazards and accidents for the Cold Vacuum Drying Facility (CVDF) is summarized in Table B5-1. This table lists TSR controls in accordance with the accident analyses in Chapter B3 0 and the safety structures, systems, and components (SSCs) evaluations in Chapter B4 0. Table B5-1 provides a road map from the respective accident analysis section to the relevant subheadings within Section B5 5, where TSR derivation details are arranged by TSR control.

The necessary and sufficient TSR controls are established based upon consideration for public safety, significant defense in depth, significant worker safety, and for maintaining radiological consequences below risk evaluation guidelines. Section 5 3 of the SNF Project FSAR contains criteria and the TSR selection process applicable to all SNF Project facilities. Section B5 3 2 contains information specific to the CVDF in addition to that provided in Section 5 3 2 of the SNF Project FSAR

B5 3 1 Criteria

The control selection criteria used for the SNF Project are described in Section 5 3 1 of the SNF Project FSAR.
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<thead>
<tr>
<th>Analyzed accident and Chapter B3 0 section</th>
<th>Technical Safety Requirement</th>
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<tbody>
<tr>
<td>Gaseous Release B3 4 2 1</td>
<td>LCO 3 3 1 MCO Vacuum Purge System Connections</td>
<td>LCO 3 3 1 provides controls to ensure leak tight connections and establish the proper vent path configuration. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td></td>
<td>LCO 3 4 1 Process General Supply/Exhaust HVAC System</td>
<td>LCO 3 4 1 provides controls to filter releases to the process bay or process water tank room that are drawn into the process general supply/exhaust HVAC system. This control is necessary to lessen the consequence of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td></td>
<td>LCO 3 4 2 Process Bay Local Exhaust HVAC and Process Vent System</td>
<td>LCO 3 4 2 provides controls to filter releases to the process bay by maintaining adequate differential pressure within the process bays. This control is necessary to lessen the consequence of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td></td>
<td>LCO 3 4 3 Reference Air System</td>
<td>LCO 3 4 3 provides controls to provide differential pressure indication and alarms. This control is necessary to lessen the consequence of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td></td>
<td>LCO 3 5 1 Diesel Generator</td>
<td>LCO 3 5 1 provides controls to provide standby power to maintain local exhaust flow and sufficient differential pressure during electrical outages. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td></td>
<td>AC 5 12 Process Bay Telescoping Door</td>
<td>AC 5 12 supports LCO 3 4 3 to maintain confinement during MCO processing.</td>
</tr>
<tr>
<td></td>
<td>AC 5 17 Dryness Testing</td>
<td>AC 5 17 provides controls to ensure adequate MCO dryness for shipment to the CSB. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td>Analyzed accident and Chapter B3 0 section</td>
<td>Technical Safety Requirement</td>
<td>Control basis</td>
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</tr>
<tr>
<td>Liquid Release B3 4 2 2</td>
<td>LCO 3 4 1 Process General Supply/Exhaust HVAC System</td>
<td>LCO 3 4 1 provides controls to filter releases to the process bay or process water tank room that are drawn into the process general supply/exhaust HVAC system. This control is necessary to lessen the consequence of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 4 3 Reference Air System</td>
<td>LCO 3 4 3 provides controls to provide differential pressure indication and alarms. This control is necessary to lessen the consequence of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td></td>
<td>AC 5 13 Combustible Loading Limits</td>
<td>AC 5 13 provides controls to keep combustible loading within the limits of the fire hazard analysis. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td>MCO External Hydrogen Explosion B3 4 2 3</td>
<td>LCO 3 1 6 Process Bay Temperature Instrumentation</td>
<td>LCO 3 1 6 provides controls to detect high bay temperatures and provide the appropriate signal to the SCIC system. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 2 1 Safety-Class Helium System</td>
<td>LCO 3 2 1 provides controls to provide a pressure regulated discharge flow path from the MCO to the local exhaust system. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 4 1 Process General Supply/Exhaust HVAC System</td>
<td>LCO 3 4 1 provides controls to filter releases to the process bay or process water tank room that are drawn into the process general supply/exhaust HVAC system. This control is necessary to lessen the consequences of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 4 2 Process Bay Local Exhaust HVAC and Process Vent System</td>
<td>LCO 3 4 2 provides controls to filter releases to the process bay by maintaining adequate differential pressure within the bays as well as maintain sufficient flow to dilute potentially hydrogen rich process gas discharges. This control is necessary to lessen both the likelihood and the consequence of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
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<td></td>
<td>LCO 3 4 3 Reference Air System</td>
<td>LCO 3 4 3 provides controls to provide differential pressure indication and alarms. This control is necessary to lessen the consequence of the accident to within the onsite risk evaluation guidelines.</td>
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<tr>
<td>MCO External Hydrogen Explosion B3 4 2 3 (continued)</td>
<td>LCO 3 4 4 High Efficiency Particulate Air Filter Loading</td>
<td>LCO 3 4 4 protects radioactive loading assumptions on HEPA filters in accident analysis</td>
</tr>
<tr>
<td></td>
<td>LCO 3 5 1 Diesel Generator</td>
<td>LCO 3 5 1 provides controls to provide standby power to maintain local exhaust flow and sufficient differential pressure during electrical outages. This control is necessary to lessen likelihood of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td></td>
<td>LCO 3 6 1 Receipt Transportation Window</td>
<td>LCO 3 6 1 protects source term and initial condition assumptions in accident analysis</td>
</tr>
<tr>
<td></td>
<td>AC 5 11 Helium Cylinder Receipt Acceptance</td>
<td>AC 5 11 provides controls to ensure improper gases are not used. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td></td>
<td>AC 5 12 Process Bay Telescoping Door</td>
<td>AC 5 12 supports LCO 3 4 3 to maintain confinement during MCO processing</td>
</tr>
<tr>
<td></td>
<td>AC 5 13 Combustible Loading Limits</td>
<td>AC 5 13 provides controls to keep combustible loading within the limits of the fire hazard analysis. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td></td>
<td>AC 5 14 Bridge Crane Movement Restrictions</td>
<td>AC 5 14 provides controls to keep the overhead crane away from key process equipment during processing except as part of an approved recovery procedure. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td></td>
<td>AC 5 17 Dryness Testing</td>
<td>AC 5 17 provides controls to ensure adequate MCO dryness for shipment to the CSB. This control is necessary to lessen likelihood of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td>Analyzed accident and Chapter B3 0 section</td>
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<td>-------------------------------------------</td>
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</tr>
<tr>
<td>MCO Internal Hydrogen Explosion B3 4 2 4</td>
<td>LCO 3 1 1 Safety-Class Instrumentation and Control System</td>
<td>LCO 3 1 1 provides controls to monitor process and facility parameters to detect and respond to abnormal conditions. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 2 Vacuum Purge System Pressure Instrumentation</td>
<td>LCO 3 1 2 provides controls to provide MCO pressure information to the SCIC system. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 3 General Service Helium System Flow Instrumentation</td>
<td>LCO 3 1 3 provides controls to provide helium flow information to the SCIC system. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 4 Tempered Water (Annulus) System Temperature Instrumentation</td>
<td>LCO 3 1 4 provides controls to detect high water supply temperatures and provide the appropriate signal to the SCIC system. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 5 Tempered Water (Annulus) System Level Detector</td>
<td>LCO 3 1 5 provides controls to detect low water levels in the cask-MCO annulus and provide the appropriate signal to the SCIC system. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 6 Process Bay Temperature Instrumentation</td>
<td>LCO 3 1 6 provides controls to detect high bay temperatures and provide the appropriate signal to the SCIC system. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 7 Seismic Trip Instrumentation</td>
<td>LCO 3 1 7 provides controls to detect and respond to seismic events. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td>Technical Safety Requirement</td>
<td>Control basis</td>
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</tr>
<tr>
<td>LCO 3.1</td>
<td>Provides controls to ensure leak tight connections and establish the proper vessel configuration. This control is necessary to lessen the likelihood of the accident to within the on-site risk evaluation guidelines.</td>
<td></td>
</tr>
<tr>
<td>LCO 2.1</td>
<td>Provides controls to purging and purge the MCQ as required to prevent flammable gas concentrations from forming. This control is necessary to lessen the likelihood of the accident to within the on-site risk evaluation guidelines.</td>
<td></td>
</tr>
<tr>
<td>LCO 3.4.1</td>
<td>Provides controls to filter releases to the process bay or process water tank room that are drawn into the process general ventilation HVAC system. This control is necessary to lessen the consequence of the accident to within the on-site risk evaluation guidelines.</td>
<td></td>
</tr>
<tr>
<td>LCO 3.4.2</td>
<td>Provides controls to filter releases to the process bay by maintaining adequate differential pressure within the process vent system, which prevent the consequence of the accident to within the on-site risk evaluation guidelines.</td>
<td></td>
</tr>
<tr>
<td>LCO 3.4.3</td>
<td>Provides controls to provide differential pressure indication and alarms. This control is necessary to lessen the consequence of the accident to within the on-site risk evaluation guidelines.</td>
<td></td>
</tr>
<tr>
<td>LCO 3.5.1</td>
<td>Provides controls to provide standby power to maintain local exhaust flow and sufficient differential pressure during electrical outages. This control is necessary to lessen the likelihood of the accident to within the on-site risk evaluation guidelines.</td>
<td></td>
</tr>
</tbody>
</table>

Table B5-1: Hazard and Accident Analyses and Technical Safety Requirement Cross Reference (11 sheets)
<table>
<thead>
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<th>Analyzed accident and Chapter B3 0 section</th>
<th>Technical Safety Requirement</th>
<th>Control basis</th>
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</thead>
<tbody>
<tr>
<td>MCO Internal Hydrogen Explosion B3 4 2 4 (continued)</td>
<td>AC 5 11 Helium Cylinder Receipt Acceptance</td>
<td>AC 5 11 provides controls to ensure improper gases are not used. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines.</td>
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<td>AC 5 12 Process Bay Telescoping Door</td>
<td>AC 5 12 supports LCO 3 4 3 to maintain confinement during MCO processing.</td>
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<td></td>
<td>AC 5 13 Combustible Loading Limits</td>
<td>AC 5 13 provides controls to keep combustible loading within the limits of the fire hazard analysis. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td></td>
<td>AC 5 14 Bridge Crane Movement Restrictions</td>
<td>AC 5 14 provides controls to keep the overhead crane away from key process equipment during processing except as part of an approved recovery procedure. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td></td>
<td>AC 5 15 MCO Process Port Valve Isolation</td>
<td>AC 5 15 provides controls to ensure that the isolation of the MCO port valves is performed in a manner such that a SCIC trip will occur if the MCO is not adequately dried. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines.</td>
</tr>
<tr>
<td>MCO Thermal Runaway Reaction B3 4 2 5</td>
<td>LCO 3 1 1 Safety-Class Instrumentation and Control System</td>
<td>LCO 3 1 1 provides controls to monitor process and facility parameters to detect and respond to abnormal conditions. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 2 Vacuum Purge System Pressure Instrumentation</td>
<td>LCO 3 1 2 provides controls to provide MCO pressure information to the SCIC system. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 4 Tempered Water (Annulus) System Temperature Instrumentation</td>
<td>LCO 3 1 4 provides controls to detect high water supply temperatures and provide the appropriate signal to the SCIC system. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td>Analyzed accident and Chapter B3 0 section</td>
<td>Technical Safety Requirement</td>
<td>Control basis</td>
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</tr>
<tr>
<td>MCO Thermal Runaway Reaction B3 4 2 5 (continued)</td>
<td>LCO 3 1 5 Tempered Water (Annulus) System Level Detector</td>
<td>LCO 3 1 5 provides controls to detect low water levels in the cask-MCO annulus and provide the appropriate signal to the SCIC system. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 6 Process Bay Temperature Instrumentation</td>
<td>LCO 3 1 6 provides controls to detect high bay temperatures and provide the appropriate signal to the SCIC system. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 7 Seismic Trip Instrumentation</td>
<td>LCO 3 1 7 provides controls to detect and respond to seismic events. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 8 Isolation Valve Interlocks</td>
<td>LCO 3 1 8 provides controls to interlock key isolation valves to prevent premature MCO draining and inadvertent water additions once dryness testing has begun. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 2 1 Safety-Class Helium System</td>
<td>LCO 3 2 1 provides controls to pressurize and purge the MCO as required to prevent flammable gas concentrations from forming. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td></td>
<td>LCO 3 3 1 MCO Vacuum Purge System Connections</td>
<td>LCO 3 3 1 provides controls to ensure leak tight connections. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
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<td></td>
<td>LCO 3 4 2 Process Bay Local Exhaust HVAC and Process Vent System</td>
<td>LCO 3 4 2 provides controls to filter releases to the process bay by maintaining adequate differential pressure within the bays as well as maintain sufficient flow to dilute potentially hydrogen rich process gas discharges. This control is necessary to lessen the consequence of the accident to within the offsite accident release limits.</td>
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</tr>
<tr>
<td>MCO Thermal Runaway Reaction B3 4 2 5 (continued)</td>
<td>LCO 3 5 1 Diesel Generator</td>
<td>LCO 3 5 1 provides controls to provide standby power to maintain local exhaust flow and sufficient differential pressure during electrical outages. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td>AC 5 10</td>
<td>Transporter Placement</td>
<td>AC 5 10 provides controls to ensure transporter placement does not result in line failure during a seismic event. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td>AC 5 11</td>
<td>Helium Cylinder Receipt Acceptance</td>
<td>AC 5 11 provides controls to ensure improper gases are not used. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td>AC 5 13</td>
<td>Combustible Loading Limits</td>
<td>AC 5 13 provides controls to keep combustible loading within the limits of the fire hazard analysis. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td>AC 5 14</td>
<td>Bridge Crane Movement Restrictions</td>
<td>AC 5 14 provides controls to keep the overhead crane away from key process equipment during processing except as part of an approved recovery procedure. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td>AC 5 15</td>
<td>MCO Process Port Valve Isolation</td>
<td>AC 5 15 provides controls to ensure that the isolation of the MCO port valves is performed in a manner such that a SCIC trip will occur if the MCO is not adequately dried. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td>AC 5 17</td>
<td>Dryness Testing</td>
<td>AC 5 17 provides controls to ensure adequate MCO dryness for shipment to the CSB. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
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</tr>
<tr>
<td>MCO Overpressurization B3 4 2 6</td>
<td>SL 2 1 1 MCO Maximum Pressure</td>
<td>SL 2 1 1 provides a maximum pressure limit for the MCO boundary</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 1 Safety-Class Instrumentation and Control System</td>
<td>LCO 3 1 1 provides controls to monitor process and facility parameters to detect and respond to abnormal conditions. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 2 Vacuum Purge System Pressure Instrumentation</td>
<td>LCO 3 1 2 provides controls to provide MCO pressure information to the SCIC system. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 4 Tempered Water (Annulus) System Temperature Instrumentation</td>
<td>LCO 3 1 4 provides controls to detect high water supply temperatures and provide the appropriate signal to the SCIC system. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 5 Tempered Water (Annulus) System Level Detector</td>
<td>LCO 3 1 5 provides controls to detect low water levels in the cask-MCO annulus and provide the appropriate signal to the SCIC system. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 6 Process Bay Temperature Instrumentation</td>
<td>LCO 3 1 6 provides controls to detect high bay temperatures and provide the appropriate signal to the SCIC system. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 7 Seismic Trip Instrumentation</td>
<td>LCO 3 1 7 provides controls to detect and respond to seismic events. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits</td>
</tr>
<tr>
<td></td>
<td>LCO 3 1 8 Isolation valve Interlocks</td>
<td>LCO 3 1 8 provides controls to interlock key isolation valves to prevent premature MCO draining and inadvertent water additions once dryness testing has begun. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits</td>
</tr>
</tbody>
</table>
# Table B5-1 Hazard and Accident Analyses and Technical Safety Requirement Cross Reference (11 sheets)

<table>
<thead>
<tr>
<th>Analyzed accident and Chapter B3 0 section</th>
<th>Technical Safety Requirement</th>
<th>Control basis</th>
</tr>
</thead>
<tbody>
<tr>
<td>MCO Overpressurization B3 4 2 6 (continued)</td>
<td>LCO 3 2 1 Safety-Class Helium System</td>
<td>LCO 3 2 1 provides controls to provide a pressure regulated discharge flow path from the MCO to the local exhaust system. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits</td>
</tr>
<tr>
<td></td>
<td>LCO 3 3 1 MCO Vacuum Purge System Connections</td>
<td>LCO 3 3 1 provides controls to ensure leak tight connections and establish the proper vent path configuration. This control is necessary to lessen the consequence of the accident to within the offsite accident release limits</td>
</tr>
<tr>
<td></td>
<td>LCS/LCO 3 3 2 Pressure Safety Relief Valves</td>
<td>LCO 3 3 2 provides controls to relieve high system pressures due to general service helium system pressure regulator failure. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td></td>
<td>LCO 3 4 1 Process General Supply/Exhaust HVAC System</td>
<td>LCO 3 4 1 provides controls to filter releases to the process bay or process water tank room that are drawn into the process general supply/exhaust HVAC system. This control is necessary to lessen the consequence of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td></td>
<td>LCO 3 4 2 Process Bay Local Exhaust HVAC and Process Vent System</td>
<td>LCO 3 4 2 provides controls to filter releases to the process bay by maintaining adequate differential pressure within the process bays. This control is necessary to lessen the consequence of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td></td>
<td>LCO 3 4 3 Reference Air System</td>
<td>LCO 3 4 3 provides controls to provide differential pressure indication and alarms. This control is necessary to lessen the consequence of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td></td>
<td>LCO 3 5 1 Diesel Generator</td>
<td>LCO 3 5 1 provides controls to provide standby power to maintain local exhaust flow and sufficient differential pressure during electrical outages. This control is necessary to lessen the likelihood of the accident to within the onsite risk evaluation guidelines</td>
</tr>
<tr>
<td></td>
<td>AC 5 10 Transporter Placement</td>
<td>AC 5 10 provides controls to ensure transporter placement does not result in line failure during a seismic event. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits</td>
</tr>
</tbody>
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Table B5-1  Hazard and Accident Analyses and Technical Safety Requirement Cross Reference  (11 sheets)

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<td>MCO Overpressurization B3 4 2 6 (continued)</td>
<td>AC 5 12 Process Bay Telescoping Door</td>
<td>AC 5 12 supports LCO 3 4 3 to maintain confinement during MCO processing</td>
</tr>
<tr>
<td></td>
<td>AC 5 13 Combustible Loading Limits</td>
<td>AC 5 13 provides controls to keep combustible loading within the limits of the fire hazard analysis. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td></td>
<td>AC 5 15 MCO Process Port Valve Isolation</td>
<td>AC 5 15 provides controls to ensure that the isolation of the MCO port valves is performed in a manner such that a SCIC trip will occur if the MCO is not adequately dried. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
<tr>
<td></td>
<td>AC 5 17 Dryness Testing</td>
<td>AC 5 17 provides controls to ensure adequate MCO dryness for shipment to the CSB. This control is necessary to lessen the likelihood of the accident to within the offsite accident release limits.</td>
</tr>
</tbody>
</table>

AC = Administrative Control  
CSB = Canister Storage Building  
HEPA = high-efficiency particulate air (filter)  
HVAC = heating, ventilation, and air conditioning  
LCO = Limiting Condition for Operation  
MCO = multi-canister overpack
B5 3 2 Safety Structures, Systems, and Components not Provided with Technical Safety Requirement Coverage

The safety-class and safety-significant SSCs discussed by section in Chapter B4 0 that have not been provided with TSR coverage are identified below. All other safety SSCs are provided with TSR coverage with a Limiting Condition for Operation (LCO) or are included in an Administrative Control (AC) program. The bases for not providing TSR coverage are provided in the following paragraphs.

B5 3 2 1 Transportation Cask The transportation cask is safety class for transportation. There are no controls for this SSC because it is a Design Feature (refer to Section B5 6) and its safety function is managed by configuration control processes. This SSC was credited as maintaining its structural integrity during the accidents.

B5 3 2 2 Multi-Canister Overpack The multi-canister overpack (MCO) is a safety-class SSC. There are no TSR controls for this safety SSC because it is a Design Feature (refer to Section B5 6) and its safety function is managed by configuration control processes. This SSC was credited as maintaining its structural integrity during the accidents.

B5 3 2 3 Cold Vacuum Drying Facility Building Structure Some functions of the CVDF structure are classified as safety-significant SSCs. There are no TSR controls for the structure because it is a Design Feature (refer to Section B5 6) and its safety function is managed by configuration control processes. The structure was not credited as a control in the analyses but was identified as a key assumption in maintaining its structural integrity during the accidents.

B5 3 2 4 Process Water Conditioning Transfer Line The process water conditioning (PWC) lines between the MCOs and the process water tank room are safety-significant SSCs. There are no TSR controls for these safety SSCs because they are Design Features (refer to Section B5 6) and their safety function is managed by configuration control processes.

B5 3 2 5 Multi-Canister Overpack and Cask Vent Jumpers The MCO and cask vent jumpers are safety-significant special tools. There are no TSR controls for these safety SSCs because they are Design Features (refer to Section B5 6) and their safety function is managed by configuration control processes.

B5 4 DERIVATION OF FACILITY MODES

B5 4 1 Operational Modes

The facility operational modes (operation, standby, and shutdown) for the CVDF were identified to indicate overall facility status. The individual process bay submodes (operation,
standby, and repair) define each process bay's status so conditions can be clearly highlighted to plant personnel.

The facility modes for the CVDF are defined as follows:

**Operation Mode** — The facility contains at least one MCO within the process bays, or the facility is administratively declared capable of receiving and/or processing MCOs although an MCO may not be present in any process bay. Maintenance activities are permitted in this mode provided the associated activities will not result in entry into the LCO action statements.

**Standby Mode** — The facility contains no MCOs within the process bays. Process water transfers to, from, or within the process water tank room may be occurring. Maintenance activities are permitted in this mode provided the associated activities will not result in entry into the LCO action statements.

**Shutdown Mode** — The facility contains no MCOs within the process bays. No process water transfers to, from, or within the process water tank room are occurring. Maintenance activities are permitted in this mode.

When the facility is in **Operation mode**, the individual process bays can be in any of the following submodes. Process bay submodes do not apply when the facility is in standby or shutdown mode.

**Operation Submode** — The process bay is receiving, containing, or processing an MCO. Maintenance activities are permitted in this submode provided the associated activities will not result in entry into the LCO action statements.

**Standby Submode** — The process bay does not contain an MCO but meets the LCO requirements necessary for it to receive an MCO. Maintenance activities are permitted in this submode provided the associated activities will not result in entry into the LCO action statements.

**Repair Submode** — The process bay does not contain an MCO and is not capable of receiving an MCO. Maintenance activities are permitted in this submode provided the associated activities will not result in entry into the LCO action statements. Note that maintenance activities involving the safety-class instrumentation and control (SCIC) system will require both of the affected process bays (i.e., those bays requiring the system's operability to meet redundancy requirements) to be in the repair submode.
B5 4 2 Minimum Staffing Requirements

The minimum operations shift complement in the Operation and Standby modes is one shift manager (or designee) and two nuclear operators. In Shutdown mode, there are no minimum staffing requirements (i.e., the facility may be unstaffed). This minimum staff in each mode is considered adequate to perform the minimum safety functions necessary to protect the health and safety of the public, onsite workers, and the environment during normal operations and abnormal and emergency conditions. The minimum staff does not necessarily include individuals necessary to fulfill the CVDF mission, goals, and objectives. The shift manager (or designee) is required to be Building Emergency Director (BED)-qualified.

Qualification training for the minimum staff (managers and nuclear operators) is addressed in Chapter 120 of the SNF Project FSAR. The program for TSR, emergency, and alarm response administrative procedures is addressed in Chapter 120. Emergency response is addressed in Chapter 150 of the SNF Project FSAR. The general considerations for determining the minimum staff are provided in Section 542 of the SNF Project FSAR, while CVDF-specific considerations are identified below.

Normal operations Determination of the minimum operations shift complement during normal operations are based on the routine surveillances required for TSR compliance. A surveillance is considered a low-difficulty task. The surveillances that must be performed during normal operations include the following:

- Verifications of process parameters such as process bay local exhaust heating, ventilation, and air conditioning (HVAC) and process vent system exhaust flow
- Verifications that equipment is in place prior to beginning specific operational tasks
- Functional tests of systems to demonstrate operability
- Instrument calibrations
- Interlock channel checks

Abnormal conditions The minimum operations shift complement during abnormal conditions is necessary to ensure compliance with required actions specified in LCO action statements with completion times of less than 8 hours or that include the term "immediately." Hanford Site experience has shown that additional staff could be provided within 8 hours, if needed (considering the most adverse weather and travel conditions), to ensure all LCO completion times are met. LCO completion times of "immediately" imply the highest sense of urgency and are given top priority over all other activities.
The minimum operations shift complement is the same in Operation and Standby modes. Specific LCO-required actions that would need to be performed immediately within a specified completion time, or prior to resuming operation (in which case a completion time is not specified), include the following:

- Manual trip of the SCIC system to provide isolation and purge of the MCO
- Verification of operating conditions (e.g., process bay local exhaust HVAC and process vent system flow)
- Restoration of system operability
- Restoration of process parameters such as tempered water (annulus) system water levels

The minimum operations shift complement is adequate during abnormal conditions to perform necessary job functions.

**Emergency conditions** The minimum operations shift complement during emergency conditions is necessary to ensure response to the spectrum of accidents analyzed in Chapter B3 (hazardous and nonhazardous). The minimum operations shift complement must make prompt initial notifications and implement initial protective actions to preclude or reduce the exposure of individuals affected by hazards or unsafe conditions during an emergency.

The shift manager (or designee) is the BED. The BED is primarily responsible for assessment of the event and protective actions at the CVDF. The BED also makes onsite notifications, implements emergency management procedures, implements facility emergency plans, classifies events, and controls event response. Two nuclear operators are considered adequate to respond to an event and support actions requested by the BED. The BED requests support services, as necessary, to perform administrative functions and the minimum functions required to ensure the health and safety of the public, onsite workers, and the environment.

DOE/RL-94-02, Hanford Emergency Management Plan, does not specify a fixed minimum staff for a particular accident at a given facility. A graded approach is used to respond to an event depending on the severity of the event. The discoverer of an event initiates response to the event. In some cases, the discoverer or other facility staff can respond adequately to the event. In other cases, resources outside the facility are required. When an emergency occurs, the event is assessed by the BED, and additional staff is obtained as necessary to respond to the accident. Two nuclear operators are considered adequate to respond to an event scene and support actions requested by the BED. Additional staff is obtained as needed.
B5 5 TECHNICAL SAFETY REQUIREMENT DERIVATION

B5 5 1 Instrumentation

B5 5 1 1 LCO 3 1 1 — Safety-Class Instrumentation and Control System

**Purpose** This control, to ensure SCIC system operability, is derived from the MCO internal hydrogen explosion accident in Section B3 4 2 4, the thermal runaway reaction accident in Section B3 4 2 5, and the MCO overpressurization accident in Section B3 4 2 6. Without controls, the release of radioactive material could exceed offsite release limits for the thermal runaway reaction and MCO overpressurization accidents, and exceed onsite risk evaluation guidelines for the internal hydrogen explosion accident. Ensuring the capability to detect and respond to process upsets (via interlocks that result in alarms, isolation of the MCO, activation of the safety-class helium system, and/or removal of power to the tempered water (annulus) system heater) reduces the risk associated with process variable perturbations.

This LCO applies to process bays in operation submode. Specific applicability for the various control functions varies according to the MCO processing steps and is consistent with the applicability of the input systems as identified in LCO 3 1 2 through LCO 3 1 7. Once the MCO is dried sufficiently (as demonstrated by AC 5 17) and isolated, MCO overpressurization thermal runaways, and hydrogen explosions are no longer a hazard for the MCO.

The LCO will require operability of the SCIC system to provide the capability to respond to process parameter upsets by providing isolation and purge of the MCO and/or removing power to the tempered water (annulus) system heater, as appropriate, and providing remote alarm indication in the control room so that further recovery actions may be taken by the operators as appropriate. Operability requirements will include the capability to receive and respond appropriately to the following inputs from process system sensors:

- Vacuum purge system (VPS) line pressure to detect both high and low MCO pressures when in pressure operations, and unexpected pressure rises (e.g., caused by inleakage) when under vacuum drying (these events will trigger an alarm, and isolation and purge of the MCO).

  **NOTE** The high MCO pressure setpoint serves as a Limiting Control Setting (LCS) to protect the MCO maximum pressure safety limit (SL 2 1 1).

- VPS line pressure to detect insufficient time at pressure or excessive time under vacuum during vacuum drying cycles (these events will trigger an alarm, and isolation and purge of the MCO).
• General-service helium system purge flow rate and VPS pressure to detect a low helium purge flow when the MCO is above 12 torr (this event will trigger an alarm, and isolation and purge of the MCO)

• Tempered water (annulus) system water level to detect low annulus water levels (this event will trigger an alarm, manual actions will be required to restore annulus water levels [LCO 3 1 5])

• Tempered water (annulus) system water temperature to detect high tempered water temperatures (this event will trigger the de-energization of the tempered water [annulus] system heater)

• Process bay temperature to detect conditions (high temperature) that may threaten the heat sink for the MCO and the safe operation of process bay instrumentation (this event will trigger an alarm, and isolation and purge of the MCO, manual actions will be required to restore process bay temperatures [LCO 3 1 6])

• Seismic activity to detect conditions that may threaten safe facility operations (this event will trigger an alarm, and isolation and purge of the MCO and de-energization of the tempered water [annulus] system heater)

Surveillances related to this LCO include periodic functional testing of system channels to demonstrate operability as well as channel checks of the interlocks for indication of the appropriate condition. Calibrations of the process system sensors that provide input to the SCIC system, although required by their associated LCO rather than this LCO, will be included with the SCIC listing of surveillances for the sake of completeness.

This control also requires LCOs 3 1 2, 3 1 3, 3 1 4, 3 1 5, and 3 1 6 to be met in order to provide the necessary sensor inputs required for the proper functioning of the system. Although the seismic trip instrumentation is considered part of the SCIC system, it has been broken out as a separate LCO (LCO 3 1 7) because it functions independently of the SCIC logic circuits.

**Derivation Criteria** This control was selected because it can be applied to multiple accidents. In addition, it serves to prevent, rather than mitigate, many of the associated accidents and uses engineered controls rather than administrative programs to achieve its safety function.

**B5 5 1 2 LCO 3 1 2 — Vacuum Purge System Pressure Instrumentation**

**Purpose** This control, to ensure VPS pressure instrumentation operability, is derived from the MCO internal hydrogen explosion accident in Section B3 4 2 4, the thermal runaway reaction accident in Section B3 4 2 5, and the MCO overpressurization accident in Section B3 4 2 6. Without controls, the release of radioactive material could exceed offsite release limits for the thermal runaway reaction and MCO overpressurization accidents, and exceed onsite risk evaluation guidelines for the internal hydrogen explosion accident.
When vacuum drying, excessive durations under vacuum or insufficient intermediate durations at pressure can place the MCO outside the analyzed envelope of existing thermal analyses for safe operations. In addition, an unexpected pressure rise during vacuum drying, as well as low MCO pressures when the MCO is in pressure operations, indicate undesirable process conditions (e.g., process system leaks or vacuum pump degradation) that could eventually lead to process upsets. The SCIC system is relied upon to respond to these conditions. Ensuring the capability to provide pressure indication to the SCIC system in order to detect vacuum cycles exceeding allowable durations, pressure operation cycles of less than allowable durations, unexpected pressure rises when vacuum–purge drying and low system pressures when in pressure operations is therefore necessary to support the operability of the SCIC system (LCO 3 1 1).

This LCO applies to process bays in operation submode from the time the MCO filtered process exit port (VPS-V-*010) is opened until such time that the final pressure rebound test (AC 5 17) is successfully completed and both MCO process port valves are closed. Once the MCO is dried sufficiently (as demonstrated by AC 5 17) and isolated, MCO overpressurization, thermal runaways, and hydrogen explosions are no longer a hazard for the CVDF.

The LCO will require operability of the VPS pressure instrumentation (PT 1*36, PT 1*37, PT 1*08, and PT 1*10) to provide the capability to sense MCO internal pressure and transmit the information to the SCIC system. Surveillances related to this LCO include periodic instrument calibrations to maintain operability.

**Derivation Criteria** This control was selected because it serves to prevent, rather than mitigate, the associated accidents and uses engineered controls rather than administrative programs to achieve its safety function.

**B5 5 1 3 LCO 3 1 3 — General-Service Helium System Flow Instrumentation**

**Purpose** This control, to ensure general-service helium system flow instrumentation operability, is derived from the MCO internal hydrogen explosion accident in Section B3 4 2 4. Without controls, the release of radioactive material could exceed onsite risk evaluation guidelines.

Low helium purge flows after completing MCO draining indicate undesirable process conditions (e.g., insufficient hydrogen dilution) that could eventually lead to process upsets. The SCIC system is relied upon to respond to these conditions. Ensuring the capability to provide flow indication to the SCIC system to detect low helium purge flows is therefore necessary to support the operability of the SCIC system (LCO 3 1 1).

This LCO applies to process bays in operation submode from the time the MCO long axial process tube port plug (VPS-V-*019) is opened until such time that the proof-of-dryness demonstration (AC 5 17) is successfully completed. At the start of the final pressure rebound...
test, the flow path through the MCO is isolated to preclude any water addition to the MCO. Once the MCO is dried sufficiently (as demonstrated by AC 5 17) (regardless of port valve position), helium flow is no longer required because of the low hydrogen generation rate within the MCO.

The LCO will require operability of the general-service helium system flow instrumentation (FIT 1*20 and FIT 1*21) to provide the capability to sense the helium flow rate and transmit the information to the SCIC system. Surveillances related to this LCO include periodic instrument calibrations to maintain operability.

**Derivation Criteria** This control was selected because it serves to prevent, rather than mitigate, the associated accidents and uses engineered controls rather than administrative programs to achieve its safety function.

**B5 5 1 4 LCO 3 1 4 — Tempered Water (Annulus) System Temperature Instrumentation**

**Purpose** This control, to ensure tempered water (annulus) system temperature instrumentation operability, is derived from the MCO internal hydrogen explosion accident in Section B3 4 2 4, the thermal runaway reaction accident in Section B3 4 2 5, and the MCO overpressurization accident in Section B3 4 2 6. Without controls, the release of radioactive material could exceed offsite release limits for the thermal runaway reaction and MCO overpressurization accidents, and exceed onsite risk evaluation guidelines for the internal hydrogen explosion accident.

High annulus water temperatures indicate undesirable process conditions in the tempered water (annulus) system (e.g., excessive heater output) that could lead to process upsets. The SCIC system is relied upon to respond to these conditions. Ensuring the capability to detect high annulus water temperatures is therefore necessary to support the operability of the SCIC system (LCO 3 1 1).

This LCO applies to process bays in operation submode when the tempered water (annulus) system supply line (TW-*01-SS-1-1/2) is connected to the lower cask process port (TW-QD-*018) and power is being supplied to the tempered water (annulus) system heater. When the supply line is connected and power is supplied to the heater, excessive heater output to the annulus water is possible. The LCO will require operability of the tempered water (annulus) system temperature instrumentation (TSH 1*28 and TSH 1*29) to provide the capability to monitor tempered water (annulus) system temperature and transmit information to the SCIC system if temperatures rise too high. Surveillances related to this LCO include periodic calibrations of the switch setpoints to maintain operability.

**Derivation Criteria** This control was selected because it serves to prevent, rather than mitigate, the associated accidents and uses engineered controls rather than administrative programs to achieve its safety function.
B5 5 1 5 LCO 3 1 5 — Tempered Water (Anulus) System Level Detector

Purpose This control, to ensure tempered water (annulus) system level detector operability, is derived from the MCO internal hydrogen explosion accident in Section B3 4 2 4, the thermal runaway reaction accident in Section B3 4 2 5 and the MCO overpressurization accident in Section B3 4 2 6 Without controls, the release of radioactive material could exceed offsite release limits for the thermal runaway reaction and MCO overpressurization accidents and exceed onsite risk evaluation guidelines for the internal hydrogen explosion accident

Low annulus water levels can, over time, lead to process upsets because of the loss of sufficient heat transfer capability The SCIC system is relied upon to alert operators to this condition so that manual corrective actions can be taken to restore tempered water (annulus) system water levels Ensuring the capability to detect low tempered water (annulus) system water levels is therefore necessary to support the operability of the SCIC system (LCO 3 1 1)

This LCO applies to process bays in operation submode from the time the tempered water (annulus) system supply line (TW-01-SS-1-1/2) is connected to the lower cask process port (TW-QD-*018) until such time that the final pressure rebound test (AC 5 17) is successfully completed Once the MCO is dried sufficiently (as demonstrated by LCO 3 2 6) (regardless of port valve position), the MCO heat generation potential is sufficiently low that annulus water loss is no longer a safety concern

The LCO will require operability of the tempered water (annulus) system level detector instrumentation (LSL 1*24 and LSL 1*25) to provide the capability to detect low annulus water levels and transmit the information to the SCIC system Manual actions will be required to restore levels to within limits Part of the required actions for this LCO will therefore include required operator response to restore adequate water levels within a specified time frame as a result of low annulus water levels Surveillances related to this LCO include periodic calibrations of the level switch to maintain operability, and verification that adequate annulus refill capabilities (i.e., special tools and water supply) are available before processing an MCO

Derivation Criteria This control was selected because tempered water (annulus) system water levels are a parameter of concern In addition, this control was selected because it serves to prevent, rather than mitigate, the associated accidents

B5 5 1 6 LCO 3 1 6 — Process Bay Temperature Instrumentation

Purpose This control, to ensure process bay temperature instrumentation operability, is derived from the MCO external hydrogen explosion accident in Section B3 4 2 3, the MCO internal hydrogen explosion accident in Section B3 4 2 4, the thermal runaway reaction accident in Section B3 4 2 5, and the MCO overpressurization accident in Section B3 4 2 6 Without controls, the release of radioactive material could exceed offsite release limits for the thermal runaway reaction and MCO overpressurization accidents, and exceed onsite risk evaluation guidelines for the internal hydrogen explosion accident
High temperatures in the process bay have the potential to exceed the rated performance limits of the safety-class pressure instrumentation. In addition, for stagnant tempered water conditions, the process bay becomes the heat sink for the MCO and must stay below 115 °F. The SCIC system is relied upon to respond to high process bay temperatures and take precautionary actions (via interlocks that actuate the safety-class helium system and alarm remotely) before performance limits are reached or maximum heat sink temperatures are exceeded. Ensuring the capability to detect high process bay temperatures is therefore necessary to support the operability of the SCIC system (LCO 311).

This LCO applies to process bays in operation submode from the time the filtered process exit port (VPS-V-*010) is opened until such time that the final pressure rebound test (AC 5 17) is successfully completed and both MCO process port valves are closed. Once the MCO is dried sufficiently (as demonstrated by AC 5 17) and isolated, the pressure instrumentation being protected by this LCO is no longer required to be operable, and the MCO is no longer at risk from high bay temperatures due to its low water content.

The LCO will require the operability of the process bay temperature instrumentation (TSHH 1*38 and TSHH 1*39) to provide the capability to detect high process bay temperatures and transmit the information to the SCIC system. Surveys related to this LCO include periodic calibrations of the temperature switch to maintain operability.

**Derivation Criteria**  
Process bay temperature control was selected because it is a parameter of concern. In addition, this control was selected because it serves to prevent, rather than mitigate, the associated accidents and uses engineered controls rather than administrative programs to achieve its safety function.

**B5 5 1 7 LCO 3 1 7 — Seismic Trip Instrumentation**

**Purpose**  
This control, to ensure seismic trip instrumentation operability, is derived from the MCO internal hydrogen explosion accident in Section B3 4 2 4, the thermal runaway reaction accident in Section B3 4 2 5, and the MCO overpressurization accident in Section B3 4 2 6. Without controls, the release of radioactive material could exceed offsite release limits for the thermal runaway reaction and MCO overpressurization accidents and exceed onsite risk evaluation guidelines for the internal hydrogen explosion accident.

Seismic events can damage the tempered water (annulus) system heater protection circuits or the MCO pressure sensor, neither of which is seismically qualified. The seismic trip instrumentation will activate the SCIC system trip independent of the logic circuits as a precaution during seismic events to perform the following actions:

- Isolate the MCO from the normal processing system
- Initiate a helium purge flow through the MCO from the safety-class helium system
- De-energize the tempered water (annulus) system heater

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Ensuring the capability to detect seismic activity is therefore necessary to support the operability of the SCIC system to perform the actions listed above (LCO 3 1 1).

This LCO applies to process bays in operation submode. The LCO will require the operability of the seismic trip instrumentation (AT-5235 AT-5336 AT-5437 including auctoneering panels CP-120 and CP-121) to detect seismic activity and trip the SCIC system in all bays processing MCOs. Surveillances related to this LCO include periodic calibrations of the three independent sensors to maintain operability.

**Derivation Criteria**  This control was selected because it serves to prevent rather than mitigate, the associated accidents and uses engineered controls rather than administrative programs to achieve its safety function.

**B5 5 1 8 LCO 3 1 8 — Isolation Valve Interlocks**

**Purpose**  This control, to ensure isolation valve interlock operability, is derived from the thermal runaway reaction accident in Section B3 4 2 5 and the MCO overpressurization accident in Section B3 4 2 6. Without controls, the release of radioactive material could exceed offsite release limits for the thermal runaway reaction and MCO overpressurization accidents.

The VPS isolation valves (on the deionized water side) prevent water from the deionized water supply from entering the MCO following MCO drying operations. The PWC isolation valves prevent water from the PWC system from entering the MCO following MCO drying operations. Inadvertent water addition to the MCO once pressure rebound testing and the dryness demonstration has begun has the potential to leave the MCO susceptible to thermal runaway reaction or overpressurization events. In addition, the PWC isolation valves ensure a premature drain of the MCO does not occur prior to placing the SCIC in the drain position.

This LCO applies to process bays in operation submode when the SCIC system is in the heatup position, as well as from the time the initial pressure rebound test begins until such time that the final pressure rebound test (AC 5 17) is successfully completed and both MCO process port valves are closed. When the SCIC system is in the heatup position, the interlock on the PWC isolation valves prevents a premature drain of the MCO when SCIC interlocks, credited while draining, are not active. During dryness testing, interlocks on both the PWC isolation valves and the VPS isolation valves (on the deionized water side) are needed to prevent water additions to the MCO once drying is complete. Once the MCO is isolated by the process port valves, water intrusion is no longer possible, and the LCO to protect against it is therefore no longer required.

The LCO will require the operability of the VPS isolation valve interlocks (on the deionized water side [VPS-GOV 1*11 and VPS-GOV 1*17]) and the PWC isolation valve interlocks (PWC-GOV 1*03 and PWC-GOV 1*30) to keep the isolation valves closed at all times when the SCIC is in the heatup position, as well as during initial pressure rebound testing, the proof-of-dryness demonstration, and final pressure rebound testing. Surveillances related to this LCO include periodic functional testing of the interlocks to ensure they are capable of performing their...
safety function  The isolation valves necessary to isolate the potential water sources must be operable to support the safety function of the interlocks. The operability of these valves is addressed in LCO 3 2 1 since these isolation valves are critical to the operability of this system as well.

**Derivation Criteria**  This control was selected because it serves to prevent, rather than mitigate, the associated accident and uses engineered controls rather than administrative programs to achieve its safety function.

**B5 5 2 Helium Systems**

**B5 5 2 1 LCO 3 2 1 — Safety-Class Helium System**

**Purpose**  This control, to ensure safety-class helium system operability, is derived from the MCO external hydrogen explosion accident in Section B3 4 2 3, the MCO internal hydrogen explosion accident in Section B3 4 2 4, the thermal runaway reaction accident in Section B3 4 2 5, and the MCO overpressurization accident in Section B3 4 2 6. Without controls, the release of radioactive material could exceed offsite release limits for the thermal runaway reaction and MCO overpressurization accidents and exceed onsite risk evaluation guidelines for the external hydrogen explosion and internal hydrogen explosion accidents.

The thermal runaway reaction accident analysis identified a situation in which the loss of the heat transfer capability normally provided by the helium within the MCO can increase the risk of a thermal runaway reaction. The hydrogen explosion accident analysis identified the need to maintain MCO headspace free of flammable concentrations of hydrogen and oxygen. The MCO overpressurization accident analysis identified a situation in which the loss of a pressure relief path from the MCO could lead to the slow pressurization of the isolated MCO and eventually to releases caused by overpressurization. Ensuring that the capability exists to correct process upsets through the activation of the safety-class helium system reduces the likelihood of these accidents.

This LCO applies to process bays in operation submode from the time the MCO long axial process tube port (VPS-V.*019) is opened until such time that the final pressure rebound test (AC 5 17) is successfully completed and both MCO process port valves are closed. Once the MCO is dried sufficiently (as demonstrated by AC 5 17) and isolated, thermal runaways, hydrogen explosions, and MCO overpressurization are no longer a hazard for the CVDF. The safety-class helium system controls consist of operability requirements to provide the capability to actuate the safety-class helium system. This includes the capability to isolate the MCO from the VPS and the PWC receiver tanks, and the capability to open connections between the MCO and the safety-class helium system to establish helium purge flow and a pressure vent path to the process bay local exhaust HVAC and process vent system. The settings of the pressure control valves ensure that venting to the process vent from the purge valves will not occur less than 60 seconds from system activation. This is important for protecting against the creation of flammable mixtures in
the process bay local exhaust HVAC and process vent system after a loss of facility power. The
delay allows time for the diesel generator to come up to power and for the process bay local
exhaust HVAC and process vent fans to come up to the maximum required flow for adequate
dilution. Surveillances related to this LCO include:

- A periodic verification that sufficient helium is present within the safety-class helium
  system to provide a 27-minute purge of the MCO to remove any air and preclude
  flammable conditions within the MCO.

- Periodic functional testing of pressure control valves to demonstrate system
  operability.

NOTE: the pressure control valve settings, in conjunction with the rupture disks
identified in Section B5 6 9, serve to protect the MCO maximum pressure safety limit
(SL 2 1 1).

- Periodic verification that the general-service helium system, VPS, and PWC system
  components (e.g., isolation valves and instrument air filters) necessary to isolate the
  MCO and maintain an intact pressure boundary for the safety-class helium system are
  capable of performing their support functions.

Manual operator action (e.g., to change out or refill helium bottles) may be necessary to
maintain this system operational in the event of a leak inside the qualified piping and valves or,
over long periods, as a part of recovery.

When the safety-class helium system is active and flowing, a minimum flow rate in the
process bay local exhaust HVAC and process vent system is required (refer to LCO 3 4 2) to
ensure flammable mixtures do not occur outside the MCO because of the mixture of hydrogen
from the MCO headspace and air in the local exhaust ventilation system.

Derivation Criteria: This control was selected because it serves to prevent, rather than
mitigate, the associated accidents and uses engineered controls rather than administrative
programs to achieve its safety function.

B5 5 3 Multi-Canister Overpack Pressure Protection

B5 5 3 1 Safety Limit 2 1 1 — Multi-Canister Overpack Maximum Pressure

Purpose: This safety limit, establishing the maximum allowable MCO pressure, is derived
from the MCO overpressurization accident in Section B3 4 2 6. Without controls, the release of
radioactive material could exceed offsite release limits. The resulting overpressurization may
result in an MCO pressure in excess of its 340 lb/in², which is the pressure a mechanically sealed
MCO can safely survive according to HNF-SD-SNF-SARR-005, *Multi-Canister Overpack Topical Report*

This safety limit applies to the process bays in Operation submode prior to the successful completion of the final pressure rebound test (AC 5 17) and the closure of both MCO process port valves (VPS-V-*010 and VPS-V-*019). Upon completion of the final pressure rebound test, the MCO is no longer capable of reaching pressures in excess of the safety limit value. Once the MCO is isolated from CVDF processing systems, it is no longer capable of being pressurized by either the normal or safety-class helium supply.

This safety limit establishes the maximum pressure an MCO at the CVDF can tolerate. Beyond this limit, primary passive barrier failure may occur if no controls are applied, with a resulting uncontrolled release to the onsite worker with unacceptable consequences. The three sources of high pressure identified in Chapter B3.0 are addressed as follows:

- LCO 3 1 1 (Safety-Class Instrumentation and Control System) provides an MCO high pressure trip setpoint that serves as the supporting LCS for this safety limit against MCO overpressurization due to isolation. In addition, LCO 3 3 1 provides additional pressure protection from MCO overpressurization by providing another controlled release path from the MCO.

- LCS/LCO 3 3 2 (Pressure Safety Relief Valves) provides the supporting LCO and LCS for this safety limit against normal helium supply overpressurization.

- In the event of safety-class helium supply overpressurization, the MCO is protected by the pressure control valves of the safety-class helium system and associated rupture disks. LCO 3 2 1 addresses the pressure control valves, while the passive design features of the rupture disks are identified in Section B5 6 9.

**Derivation Criteria.** Although this pressure parameter does not meet all the criteria identified within Section 5 3 1 2 of the SNF Project FSAR required for selection of this parameter as a safety limit, it was deemed necessary by the U.S. Department of Energy, Richland Operations Office. It is therefore included in order to define acceptable pressure values for the CVDF. It is important to note, however, that the MCO is capable of being isolated from CVDF pressure instrumentation systems during processing. In this configuration, the MCO is capable of exceeding pressure limits, even though the pressure is neither under operator control nor directly measurable at these times. While controls are credited in Chapter B3 0 and applied with the TSRs to prevent excessive pressures in this configuration, the parameter does not conform to the typical convention of a safety limit.
B5 5 3 2 LCO 3 3 1 — Multi-Canister Overpack Vacuum Purge System Connections

**Purpose** This control, to ensure MCO VPS connections are properly made and proper confinement boundaries are established, is derived from the gaseous release accident in Section B3 4 2 1, the MCO internal explosion accident in Section B3 4 2 4, the thermal runaway reaction accident in Section B3 4 2 5, and the MCO overpressurization accident in Section B3 4 2 6. Without controls, the release of radioactive material could exceed offsite release limits for the thermal runaway reaction accident and the MCO overpressurization accident, and exceed onsite risk evaluation guidelines for the gaseous release accident and the MCO internal hydrogen explosion accident. Functionality of the MCO vent path and VPS connection interfaces provides a method of both ensuring a controlled release from the MCO in overpressurization situations, and preventing leakage from the MCO caused by faulty process connections.

This LCO applies to process bays in Operation submode from the time the filtered process exit port (VPS-V-*010) is opened until such time that the final pressure rebound test (AC 5 17) is successfully completed and both MCO process port valves are closed. Once the MCO is properly dried and isolated, pressure control from the 30 lb/in² gauge VPS vent path (VPS-PSE-1*33) is no longer needed, and system leakage no longer is a hazard for the MCO. This LCO consists of requirements to properly establish the MCO process connections to relieve high MCO pressures in a controlled manner. Surveillances related to this LCO include the following:

- Verifications while making process connections that the vent path is aligned properly, this ensures that overpressurization protection is in place for MCO processing.

- VPS system process connections (VPS-V-*010 and VPS-V-*019) are verified to be leak-tight (performed by a vacuum decay test or by pressurizing the MCO long axial process tube port and process vent port with helium and using mass spectrometer leak detection capability or other available means to detect any helium leakage), this protects safety analysis assumptions in relation to air inleakage into the MCO.

- Periodic verifications that the active components of the 30 lb/in² gauge vent path (VPS-PSE-1*33) are capable of performing their safety function (e.g., that the spring-loaded pressure relief valve [VPS-CKV-*112]) cracks open at pressures ≤35 lb/in² gauge and reseats at pressures ≥20 lb/in² gauge), these settings ensure that any hydrogen vented to the process bay local exhaust HVAC and process vent system does not result in the creation of flammable mixtures.

**Derivation Criteria** This control was selected because the VPS connection points are part of the isolation boundary of the MCO, making this control necessary for a wide range of accident scenarios (e.g., those involving isolation of the MCO). Once connected to the MCO, the MCO 30 lb/in² gauge vent path actually becomes part of the primary confinement boundary.
B5 5 3 3 LCS/LCO 3 3 2 — Pressure Safety Relief Valves

Purpose This control, to ensure general-service helium system safety relief valve operability, is derived from the MCO overpressurization accident in Section B3 4 2 6. Without controls, the release of radioactive material could exceed offsite release limits for this accident. Ensuring the operability of these valves reduces the likelihood of this accident to within the offsite release limits and onsite risk evaluation guidelines.

This LCS/LCO applies to the facility in Operation mode. The general-service helium system safety relief valve controls consist of operability requirements that ensure that helium supply pressures are reduced to acceptable levels prior to the helium supply entering the process system. The setpoint of the pressure safety valves (He-SRV-5026 and He-SRV-5031) is <30 lb/in² gauge and serves as an LCS to protect the safety limit. Surveillances related to this LCS/LCO include functional testing of the pressure safety relief valves to verify operability.

Derivation Criteria This control was selected because it serves to prevent, rather than mitigate, the associated accidents and uses engineered controls rather than administrative programs to achieve its safety function.

B5 5 4 Heating, Ventilation, and Air Conditioning Systems

B5 5 4 1 LCO 3 4 1 — Process General Supply/Exhaust Heating, Ventilation, and Air Conditioning System

Purpose This control, to ensure process general supply/exhaust HVAC system operability, is derived from the gaseous release accident in Section B3 4 2 1, the liquid release accident in Section B3 4 2 2, the MCO external hydrogen explosion accident in Section B3 4 2 3, the MCO internal hydrogen explosion accident in Section B3 4 2 4, and the MCO overpressurization accident in Section B3 4 2 6. Without HVAC controls, the release of radioactive material could exceed onsite risk evaluation guidelines for these accidents. Ensuring the operability of this system reduces the consequences of these accidents to within the onsite risk evaluation guidelines.

The process general supply/exhaust HVAC system high-efficiency particulate air (HEPA) filtration is required to accommodate the presence of general ventilation flow within the bay at the time of a postulated event, drawing radioactive material through the system. This LCO applies when the facility is in operation and standby modes when the process general supply/exhaust HVAC system fans are operating. When the fans are not operating, no differential pressure is being provided by this system, so filtration requirements no longer apply. The process general supply/exhaust HVAC system controls consist of an operability requirement that the exhaust HEPA filter provide a testable HEPA filtration of no less than 99.9%. Surveillances related to this LCO include an annual aerosol test of efficiency to demonstrate operability.
Derivation Criteria  This control was selected because it serves to mitigate multiple accidents to minimize implementation cost

B5 5 4 2 LCO 3 4 2 — Process Bay Local Exhaust Heating, Ventilation, and Air Conditioning and Process Vent System

Purpose  This control, to ensure process bay local exhaust HVAC and process vent system operability, is derived from the gaseous release accident in Section B3 4 2 1, the MCO external hydrogen explosion accident in Section B3 4 2 3, the MCO internal hydrogen explosion accident in Section B3 4 2 4, the thermal runaway reaction accident in Section B3 4 2 5, and the MCO overpressurization accident in Section B3 4 2 6. Without HVAC controls, the release of radioactive material could exceed onsite risk evaluation guidelines for these accidents. Ensuring the operability of this system reduces the likelihood and consequence of these accidents to within the onsite risk evaluation guidelines.

The process bay local exhaust HVAC and process vent system is required to provide a reliable source of differential pressure (i.e., confinement) between the process bays and the environment, and to filter any releases to the process vent. The process bay local exhaust HVAC and process vent system controls consist of operability requirements that include the following elements. The elements apply for process bays in Operation submode from the time the process bay doors are closed until such time that the final pressure rebound test (AC 5 17) is successfully completed and the MCO process port valves are closed (Flow switch interlock requirements only apply when cask venting is occurring).

- A minimum flow rate of 1,000 standard ft³/min shall be maintained through the system (as adjusted for instrumentation error). This ensures confinement and prevents flammable mixtures of air and hydrogen in the process bay local exhaust HVAC and process vent system. Operability requirements include the instrumentation necessary to verify flow (FS-8*07) and remote alarms to alert facility personnel when this minimum flow is not met.

- When the process bay doors are closed, the portion of the system between (and including) the process hood isolation dampers, the exhaust fans, and the safety-class helium system must be intact and must provide a testable HEPA filtration of no less than 99.9%. This ensures the capability to adequately filter particulate release from the MCO, both during normal flow states (should an accidental release occur while normal flow is active) and during safety-class helium system purging.

- The instrument air system piping, tank, and check valve (PI-5*20) shall be leak tight and have a tank pressure of ≥90 lb/in². This ensures the capability exists to open the hood isolation damper as part of system restart during a loss of power.

- A flow switch interlock (FS-8*52) shall be present and operable to ensure safe cask and MCO venting. The interlock shall close the flow valve when process bay local...
HVAC and process vent system flows drop below 1,000 standard ft\(^3\)/min (as adjusted for instrumentation error), effectively terminating the venting operation until proper flows can be reestablished.

Surveillances related to this LCO include annual aerosol testing of efficiency, as well as instrumentation and setpoint calibrations and functional checks. The system will be functionality tested periodically to ensure 1,000 standard ft\(^3\)/min (as adjusted for instrumentation error) can be achieved within 30 seconds of receiving diesel power (Diesel power must be supplied within 30 seconds of power loss to meet the overall requirement to establish the minimum flow within 60 seconds of facility power loss.) This is necessary to ensure adequate flows are present in the process bay local exhaust HVAC and process vent system before safety-class helium purge flow vents to the system.

**Derivation Criteria** This control was selected because the local exhaust ventilation system serves to mitigate multiple accidents to minimize implementation cost.

**B5 4 3 LCO 3 4 3 — Reference Air System**

**Purpose** This control, to ensure reference air system differential pressure instrumentation operability, is derived from the gaseous release accident in Section B3 4 2 1, the liquid release accident in Section B3 4 2 2, the MCO external hydrogen explosion accident in Section B3 4 2 3, the MCO internal hydrogen explosion accident in Section B3 4 2 4, and the MCO overpressurization accident in Section B3 4 2 6. Without HVAC controls, the release of radioactive material could exceed onsite risk evaluation guidelines for these accidents. The operability of this system plays a role in reducing the consequences of these accidents to within the onsite risk evaluation guidelines.

The differential pressure instrumentation of the reference air system is necessary to alert facility personnel to a loss of confinement being provided by the balance of general, local, and recirculation HVAC systems. The reference air system controls consist of operability requirements that include the following elements:

- Instrumentation (PDI-8*20 and PDI-8080) necessary to measure differential pressures between the area of concern (process bay or process water tank room) and the reference air system (a relative negative pressure [i.e., confinement] must be maintained within the areas of concern in order for the process bay's confinement function to be considered met).

- Remote differential pressure alarm instrumentation indicating the loss of confinement.

This LCO applies when the facility is in operation and standby modes. Individual process bay confinement monitoring and alarm requirements are applicable only if an MCO is present within the bay and the process bay door is closed. Process water tank room confinement monitoring and alarm requirements are applicable only if the circulation pump is running.
Surveillances related to this LCO include periodic calibrations of the differential pressure instrumentation and functional checks of the remote alarm.

**Derivation Criteria**

This control was selected because it serves to mitigate multiple accidents to minimize implementation cost.

**B5 5 4 4 LCO 3 4 4 — High-Efficiency Particulate Air Filter Loading**

**Purpose**

This control derives from the MCO external hydrogen explosion accident in Section B3 4 2 3. The purpose of this control is to limit the radioactive inventory available for release from the HEPA filters contained in the process bay local exhaust HVAC and process vent system based on the results of the analysis, the release of radioactive material without controls could exceed onsite risk evaluation guidelines. This LCO will require that HEPA filter and prefilter housing contact dose radiation levels be maintained equal to or less than 82 mR/h consistent with the source term assumed within the accident analysis to ensure that controls identified for the risk reduction of this accident remain valid by limiting the inventory on the filters available for release.

This LCO applies to the facility in operation mode. In other facility modes, there are no sources of hydrogen present within the facility to cause a hazard associated with explosions in the process bay local exhaust HVAC and process vent system. Surveillances related to this LCO include periodic checks of HEPA filter housing contact dose rates.

**Derivation Criteria**

This control protects an assumption within the analysis relative to filter loading available for release.

**B5 5 5 Standby Power**

**B5 5 5 1 LCO 3 5 1 — Diesel Generator**

**Purpose**

This control, to ensure diesel operability, is derived from the gaseous release accident in Section B3 4 2 1, the MCO external hydrogen explosion accident in Section B3 4 2 3, the MCO internal hydrogen explosion accident in Section B3 4 2 4, the thermal runaway reaction accident in Section B3 4 2 5, and the MCO overpressurization accident in Section B3 4 2 6. Without controls, the release of radioactive material could exceed offsite release limits for the thermal runaway reaction and MCO overpressurization accidents, and exceed onsite risk evaluation guidelines for the gaseous release, MCO external hydrogen explosion, and MCO internal hydrogen explosion accidents.

Upon loss of facility power, the diesel generator will be relied upon to provide power to the local exhaust fans and isolation damper actuator solenoids such that satisfactory local exhaust flow is maintained. This flow is necessary both to dilute hydrogen releases to the local exhaust from the MCO, as well as to maintain adequate differential pressure within the process bays.
Without differential pressure detection (because of the power loss), a configuration is required that ensures an adequate differential pressure. A local exhaust restart circuit will place the process bay local exhaust HVAC and process vent system in the required configuration (including fan motor logic) to ensure adequate system flow within 60 seconds after a loss of facility power. (Diesel power must be supplied to the process bay local exhaust HVAC and process vent system within 30 seconds, and the process bay local exhaust HVAC and process vent system fans must come up to the necessary speed within an additional 30 seconds.)

This LCO applies when the facility is in Operation mode. The LCO will require operability of the diesel generator to provide power to the local exhaust HVAC and process vent system within 30 seconds of loss of offsite power. Surveillances related to this LCO will include the following:

- Periodic verifications of sufficient diesel fuel volume
- Periodic verification of fuel quality that could affect its performance
- Periodic inspections of the starting air system
- Periodic functional testing of the diesel generator including the local exhaust restart circuit to demonstrate that the diesel generator will automatically start when required and is capable of assuming the required loads

**Derivation Criteria**  This control was selected because power is a support function required for the continued operability of the process bay local exhaust HVAC and process vent system flow. The diesel was broken out as a separate control in order to highlight its critical function in the maintenance of continued HVAC flow for differential pressure and hydrogen mitigation and to provide a means for identifying the specific requirements for required actions upon loss of diesel availability.

**B5 5 6  Transportation-Related Activities**

**B5 5 6 1  LCO 36 1 — Receipt Transportation Window**

**Purpose**  This control derives from the MCO external hydrogen explosion accident in Section B3 4 2 3. The purpose of this control is to limit the maximum cask–MCO receipt pressure to within the initial condition assumptions of the analysis.

This LCO applies to the facility in Operation mode. This LCO will require that the transportation cask be vented to the process bay local exhaust HVAC and process vent system within 24 hours of completing the helium purge of the MCO headspace at the K Basins. Required actions will identify any additional precautionary measures needed for the handling of out-of-
specification MCOs at the CVDF. A surveillance to verify the venting of the cask within the transportation window time limit will be part of this LCO.

**Derivation Criteria.** This control was selected to protect initial condition assumptions relative to maximum cask-MCO receipt pressure and hydrogen concentrations.

**B5 5 7 Administrative Controls**

**B5 5 7 1 AC 5 7 — Nuclear Criticality Safety**

**Purpose.** This control will protect the assumptions of the nuclear criticality evaluation in Chapter B6 0 to ensure that CVDF operations minimize the risk of nuclear criticality. Key elements of this programmatic AC are derived from contractor procedures that include requirements for criticality safety evaluations, criticality prevention specifications, and criticality training. The criticality controls identified in Chapter B6 0 are derived from HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*, and HNF-SD-SNF-CSER-006, *Criticality Safety Evaluation Report for the Cold Vacuum Drying Facility's Process Water Handling System*.

**Derivation Criteria.** A nuclear criticality program is required according to DOE Order 5480 22, *Technical Safety Requirements*, Section 9 e (5) as one of the standard ACs.

**B5 5 7 2 AC 5 8 — Measurement and Test Equipment**

**Purpose.** This control provides a support control for the measuring and test equipment used to verify process parameters to comply with the TSRs when specific instrumentation and associated surveillances are not included in the TSR. This control applies to (1) optional instrument systems allowable in accordance with the LCO bases, and (2) instrumentation used to verify parameters for the ACs in the TSR. This control on the measuring and test equipment ensures that process parameters are properly monitored.

The program key elements include maintaining identification and traceability of TSR-related instrumentation and equipment, periodic testing of the instrumentation and equipment used, and maintaining records that demonstrate that the instrumentation and equipment were used while in calibration.

The program applies to installed and portable measuring and test equipment when used to verify process parameters specified in the TSRs where specific instrumentation and associated surveillances are not already identified in the TSR. This program does not apply to equipment such as rulers, tape measures, levels, and other measurement devices if commercial equipment provides accuracy adequate to meet the measurement specifications.
Annex B - Cold Vacuum Drying Facility

Derivation Criteria: This control was included because for some process parameters in the TSR, it is not essential to specify the particular instrumentation or measuring equipment used to demonstrate compliance with the parameter involved provided the instrumentation or equipment used to monitor the parameter is maintained. This support control was selected because it provides operational flexibility to use various instrumentation and equipment to measure TSR parameters provided it complies with this AC while ensuring the safety functions assumed in the accident analysis are met.

B5 5 7 3 AC 5 9 — Configuration Management

Purpose: This configuration management AC was deemed necessary to protect the safe operation of the facility against uncontrolled changes to Design Features. A sitewide institutional safety program for configuration management is in place to control such changes. In addition, the Spent Nuclear Fuel Project has developed its own plan, HNF-SD-SNF-CM-001, Spent Nuclear Fuel Project Configuration Management Plan, to handle configuration management at the Spent Nuclear Fuel Project facilities. The plan defines the process that the Spent Nuclear Fuel Project has approved to secure configuration management and it is formatted to include the elements and functions established in DOE-STD-1073-93, Guide for Operational Configuration Management Program. The plan includes compliance requirements with the overall Site configuration management system and criteria for change control, document control, configuration management implementation, and periodic assessments. This AC heightens attention to the need for CVDF compliance with these configuration management plans.

Derivation Criteria: This control was deemed necessary by the Project Hanford Management Contract Contractor to protect the safe operation of the facility against uncontrolled changes to Design Features. HNF-PRO-700, Safety Analysis and Technical Safety Requirements, states that configuration management is a candidate AC when Design Features have been identified for a facility.

B5 5 7 4 AC 5 10 — Transporter Placement

Purpose: The transporter placement control is derived from the accident analysis for an MCO internal hydrogen explosion in Section B3 4 2 4, the accident analysis for a thermal runaway reaction in Section B3 4 2 5, and the accident analysis for MCO overpressurization in Section B3 4 2 6. Without controls, the thermal runaway reaction and MCO overpressurization accidents have consequences that could exceed offsite release limits, and the internal hydrogen explosion accident has consequences that could exceed onsite risk evaluation guidelines.

This control will require that a program be established and maintained to ensure that before connecting CVDF systems to the MCO, the cask, which is located upon the trailer, is positioned such that the cask is no greater than an established safe distance from the ideal horizontal placement, as identified in Chapter B4 0. This control protects the MCO during seismic events. Seismic calculations indicate that in the event of an earthquake, the transporter movement will not
be sufficient to damage the key process lines or equipment required to maintain the MCO in a safe configuration provided the cask location is within a specified distance of the ideal

**Derivation Criteria**  This control protects an assumption within the safety analysis regarding limits on expected seismic responses

### B5 5 7 5  AC 5 11 — Helium Cylinder Receipt Acceptance

**Purpose**  The helium cylinder receipt acceptance control is derived from the accident analysis for an MCO external hydrogen explosion in Section B3 4 2 3, for an MCO internal hydrogen explosion in Section B3 4 2 4, and for a thermal runaway reaction in Section B3 4 2 5. Without controls, the thermal runaway reaction accident has consequences that could exceed offsite release limits, and the external hydrogen explosion and internal hydrogen explosion accidents have consequences that could exceed onsite risk evaluation guidelines.

This control will require that a program be established and maintained to ensure that the manufacturer's paperwork and shipping papers are checked upon helium cylinder receipt at the CVDF, both the normal and safety-class supply, to verify that the cylinder's contents were sampled by the supplier and that the sample met the required purity specification of >99% helium. This control, in conjunction with the facility's institutional quality assurance program, protects against impurities or improper gases being fed by the normal or safety-class helium gas supply into the MCO during processing operations.

**Derivation Criteria**  This control protects an assumption within the safety analysis regarding helium supply purity. Sampling the cylinder's contents reduces the likelihood of connecting helium cylinders into the helium system with contents that could lead to accident conditions.

### B5 5 7 6  AC 5 12 — Process Bay Telescoping Door

**Purpose**  The process bay telescoping door control supports LCO 3 4 3, "Reference Air System," for the gaseous release accident in Section B3 4 2 1, the accident analysis for an MCO external hydrogen explosion in Section B3 4 2 3, for an MCO internal hydrogen explosion in Section B3 4 2 4, and for an MCO overpressurization accident in Section B3 4 2 6. Without HVAC confinement controls, the onsite consequences of these accidents could exceed onsite radiological risk evaluation guidelines.

This control will require that a program be established and maintained to ensure that the process bay telescoping door be closed before connecting any CVDF process systems to the MCO. Opening the telescoping door during processing has the potential to negate the confinement function within the process bay that the various HVAC systems provide. The telescoping door may not be opened again until all process connections have been removed from the MCO except when such action is required as part of an approved recovery plan.
**Derivation Criteria** This control was selected to protect the confinement function of the process bay local exhaust HVAC and process vent system during processing operations.

**B5 57 7 AC 513 — Combustible Loading Limits**

**Purpose** The combustible loading limits control is derived from the liquid release accident in Section B3 4 2 2, the MCO external hydrogen explosion accident in Section B3 4 2 3, the MCO internal hydrogen explosion accident in Section B3 4 2 4, the thermal runaway reaction accident in Section B3 4 2 5, and the MCO overpressurization accident in Section B3 4 2 6. This control is established based on Chapter B3 0 assumptions made regarding combustible loading limits in relation to the fire hazard implementation plan (SNF-4942).

This control applies in the facility Operation and Standby modes. A program shall be established and maintained to limit the combustible loadings as determined by the fire hazard implementation plan (SNF-4942).

**Derivation Criteria** This control was selected to protect Chapter B3 0 assumptions relating to the fire hazard analysis. Without this control, the safety analysis conclusion that no release of radioactive material to the environment will result due to the evaluated fire scenarios could be invalidated.

**B5 57 8 AC 514 — Bridge Crane Movement Restrictions**

**Purpose** The bridge crane movement restriction control is derived from the MCO external hydrogen explosion accident in Section B3 4 2 3, the MCO internal hydrogen explosion accident in Section B3 4 2 4, and the thermal runaway reaction accident in Section B3 4 2 5. Without controls, the thermal runaway reaction accident has consequences that could exceed offsite release limits, and the external and internal hydrogen explosion accidents have consequences that could exceed onsite risk evaluation guidelines.

This control applies to process bays in the Operation submode. This control will require that a program be established and maintained to ensure that overhead bridge crane movement is not allowed from the time the tempered water (annulus) system supply line (TW-*01-SS-1-1/2) is connected to the lower cask process port (TW-QD-*018) until such time as the final pressure rebound test is successfully completed and both MCO process port valves (VPS-V-*010 and VPS-V-*019) are closed (except when such action is required as part of an approved recovery plan). Bridge crane movement has the capability to shear key process lines and lead to unacceptable accident conditions. Restricting bridge crane movement when the MCO is susceptible to external and internal hydrogen explosions and thermal runaway reactions reduces the likelihood of these events.

**Derivation Criteria** This control was selected because it serves to prevent, rather than mitigate the associated accidents.
B5 5 7 9 AC 5 15 — Multi-Canister Overpack Process Port Valve Isolation

**Purpose** The MCO process port valve isolation control is derived from the MCO internal hydrogen explosion accident in Section B3 4 2 4, the MCO thermal runaway reaction accident in Section B3 4 2 5, and the MCO overpressurization accident in Section B3 4 2 6. Without controls, the thermal runaway reaction and high pressure blowdown have consequences that could exceed offsite release limits for the thermal runaway reaction and MCO overpressurization accidents, and could exceed onsite risk evaluation guidelines for the MCO internal hydrogen explosion accident.

This control applies in the process bay Operation submode. This control will require that a program be established and maintained to ensure that when isolating the MCO with the process port valves, the long axial process tube port (VPS-V-*019) must be closed at least 5 minutes prior to closing the filtered process exit port (VPS-V-*010). This will ensure that an MCO still processing will initiate an SCIC trip during the MCO isolation steps. This SCIC trip serves as a means of detecting human errors in which the incorrect bay was entered for MCO shipment preparation, prior to the MCO actually meeting shipment criteria.

**Derivation Criteria** This control was selected because it serves to prevent, rather than mitigate, the associated accident. The human error of mistakenly entering the wrong bay for MCO isolation has the capability to defeat most of the engineered features of the CVDF. This is because the premise of MCO isolation is that disconnection from the process systems which protect the MCO is an acceptable action to take. By requiring the proper process port isolation sequence and delay as part of the MCO isolation process, the likelihood of discovery of the error is increased significantly. This program ensures that control room alarms that are not expected during the disconnection process occur for MCOs not meeting dryness criteria.

B5 5 7 10 AC 5 16 — Preparing the Multi-Canister Overpack for Shipment

**Purpose** The control for preparing an MCO for shipment is derived from the assumptions in the safety analyses in Annex A, the Canister Storage Building (CSB) FSAR. A maximum MCO leakage rate of less than $10^{-5}$ standard cm$^3$/s is assumed in the analyses to prevent unacceptable onsite consequences caused by external hydrogen explosions.

This control applies to process bays in the Operation submode. This control will require that a program be established and maintained to ensure that the MCO mechanical seal and cover plate seals are leak tested to confirm seal integrity. Shipment of the MCO to the CSB is not allowed until an MCO leakage rate of less than $10^{-5}$ standard cm$^3$/s is demonstrated.

**Derivation Criteria** This control was selected to protect safety analysis assumptions in the safety analyses in Annex A, the CSB FSAR, and serves to prevent, rather than mitigate, the associated CSB accidents.
B5 5 7 11 AC 5 17 — Dryness Testing

**Purpose** This AC derives from the gaseous release accident in Section B3 4 2 1, the MCO external hydrogen explosion accident in Section B3 4 2 3, the thermal runaway reaction accident in Section B3 4 2 5, and the MCO overpressurization accident in Section B3 4 2 6. Without controls, the release of radioactive material could exceed offsite release limits for the thermal runaway reaction accident and MCO overpressurization accident, and exceed onsite risk evaluation guidelines for the gaseous release accident and the MCO external hydrogen explosion accident. The dryness tests conducted at the CVDF demonstrate adequate dryness levels within an MCO to proceed to the next processing step and, finally, to proceed with shipment preparation. Ensuring that each of these critical decision steps is adequately performed prevents hazards associated with excessive MCO water content following vacuum drying.

This AC applies to process bays in Operation submode. This AC will require meeting an initial pressure rebound test surveillance (pressure rise test) before entry into the proof-of-dryness demonstration is allowed. Similarly, a proof-of-dryness demonstration surveillance must be met before the final pressure rebound test steps can begin. Finally, a final pressure rebound test must be met before shipment preparation steps that involve MCO reconfiguration can begin.

These control requirements are established to prevent accidents at the CVDF related to excessive MCO water content following vacuum drying. In addition, they protect an interface between the CVDF and CSB. Dryness testing is critical for ensuring the CVDF mission has been successful (i.e., the applicable CSB receipt acceptance criteria are met) and the MCO may safely be shipped to the CSB.

**Derivation Criteria** This control was selected to protect safety analysis assumptions within the CVDF and CSB FSAR Annexes. It serves to prevent, rather than mitigate, the associated accidents.

B5 5 7 12 AC 5 18 — Safety Programs

**Purpose** This AC provides commitment to the following Environment, Safety, and Health Organization programs:

- In-service surveillance and maintenance
- Procedures and training
- Quality assurance

**Derivation Criteria** The commitment to an in-service surveillance and maintenance program is based on the complexity of the safety equipment relied on for prevention of accidents. The commitment to a procedures and training program is based on the high level of dependence placed on operator actions. The commitment to a quality assurance program is based on the need to ensure vendors’ quality assurance programs (important for ensuring cylinder gas purity) is adequate.
B5 6 DESIGN FEATURES

Design Features for the CVDF that, if altered or modified, would have a significant effect on safe operation are listed below. Descriptions of these Design Features are provided in Chapter B2. The safety functions they perform are provided in Chapter B4.

B5 6.1 Process Bay Local Exhaust Heating, Ventilation, and Air Conditioning and Process Vent System

Design Features for the process bay local exhaust HVAC and process vent system include the following:

- The ductwork from the fail-closed process hood isolation dampers, the safety-class helium system, and the 30 lb/in² gauge vent path must route discharge air to the process bay local exhaust HVAC and process vent system HEPA filter.
- The MCO and cask vent jumpers shall direct the flow of hydrogen through an orifice device located in the process bay local exhaust HVAC and process vent system.
- The orifice device to restrict the flow rate of hydrogen into the process bay local exhaust HVAC and process vent system. The orifice diameter must be sufficiently small such that the minimum process bay local exhaust HVAC and process vent system flow rate will dilute hydrogen concentrations below the lower flammability limit.

B5 6.2 Tempered Water (Annulus) Water System

Design Features for the tempered water (annulus) system include the following:

- The lines from the cask must rise above the minimum acceptable water level in the annulus.
- The lines identified above must be seismically qualified and have antisiphon devices to prevent draining of the cask annulus (see LCO 3 1 5 for surveillance and verification of annulus refill capabilities).
- The system must have the appropriate piping and refill port to accommodate a manual fill of the cask annulus with water.
B5 6.3 Transportation Cask

Design Features for the transportation cask include the following:

- The transportation cask must provide a pressure boundary prior to cask venting.
- The transportation cask must be dimensionally stable and provide a boundary to maintain adequate annulus water levels in conjunction with the tempered water (annulus) system.
- The transportation cask must provide adequate personnel shielding during process operations.
- The cask drain fitting must retain its integrity as a leak-tight connection to the tempered water supply piping.

B5 6.4 Multi-Canister Overpack

Design Features for the MCO include the following:

- The MCO must provide primary confinement for the fuel elements and fuel fragments.
- The MCO must provide a pressure boundary for the safety-class helium system.
- The MCO shell and closure plug, including the internal filter guard, must have a geometrically favorable design to prevent criticality.
- The MCO Mark IA fuel and scrap baskets’ center posts and baseplates must have a geometrically favorable design to prevent criticality.
- Any MCO scrap baskets must have flow bypass restricters (wipers) and copper inserts adequate to ensure the heat transfer in the MCO remains consistent with thermal analysis assumptions.
- The MCO shield plug must provide adequate personnel shielding during process operations.
- The MCO shield plug must have an active rupture disk with a setting of 150 lb/in² during CVDF processing.
Annex B — Cold Vacuum Drying Facility

B5 6 5 Cold Vacuum Drying Facility Building Structures

Design Features for the CVDF building structures include the following

- The process bays and the process water tank room floor, slab, walls, and roof, in conjunction with the process general supply/exhaust HVAC system, must provide confinement for gaseous releases

- The structural supports of the process bays must maintain structural integrity throughout all design basis natural phenomena hazard events (performance category 3)

- The structural supports of the process water tank room and process support area (including the second floor mechanical equipment room) must maintain structural integrity throughout all design basis natural phenomena hazard events (performance category 2)

- The CVDF exhaust stack must maintain structural integrity throughout all design basis natural phenomena hazard events (performance category 2)

- The remaining structure (administrative area) must be designed such that should it fail because of a seismic event, it fail in a manner that does not impact the seismically qualified portions

- Fire-rated structures identified in the fire hazard analysis (SNF-4268) and fire hazard implementation plan (SNF-4942) must provide the necessary fire protection characteristics as specified within the analysis

B5 6 6 Process Water Conditioning Transfer Line

Design Features for the PWC transfer line between the MCO and the process water tank room include the following

- The line height must be at least 8 ft above the floor to protect against vehicle impacts within the bay

- The line must be qualified to withstand all anticipated environmental factors (e.g., corrosion)
B5 6 7 Multi-Canister Overpack and Cask Vent Jumpers

Design Features for the MCO and cask vent jumpers include the following

- The MCO and cask vent jumpers must route discharge gas to the process bay local exhaust HVAC and process vent system

B5 6 8 Reference Air System

Design Features for the reference air system include the following

- The reference air system must provide a common reference point for comparisons of differential pressure within the facility

B5 6 9 Safety Class Helium System

Design Features for the safety class helium system include the following

- Each leg of the safety class helium system must contain an active rupture disk with a setting of 125 lb/in^2 gauge during CVDF processing

B5 6 10 Process Bay Recirculation Heating, Ventilation, and Air Conditioning System

Design Features for the process bay recirculation HVAC system include the following

- The supply dampers of the process bay recirculation HVAC system must fail closed upon a loss of power

B5 6 11 Process General Supply/Exhaust Heating, Ventilation, and Air Conditioning System

Design Features for the process general supply/exhaust HVAC system include the following

- The pneumatic isolation dampers for the process bays must fail closed upon a loss of power or instrument air

- The ductwork from the process bays and the process water tank room must route discharge air to the process general supply/exhaust HVAC system HEPA filter
B5 6 12 Vacuum Purge System

Design Features for the VPS include the following

- The MCO primary vent path must contain an active rupture disk with a setting of 30 lb/in² gauge during CVDF processing

B5 7 INTERFACES WITH TECHNICAL SAFETY REQUIREMENTS FROM OTHER FACILITIES

The TSRs of other facilities that affect the SNF Project facilities safety basis are briefly summarized in Section 5.7 of the SNF Project FSAR. TSRs of other facilities that affect the CVDF are summarized in the following subsections

B5 7 5 Receipt and Transfer of Spent Nuclear Fuel

The CVDF interfaces with other facilities through the receipt of SNF from the K Basins and the transfer of the dried fuel to the CSB. This section describes the SNF material transfers to and from the CVDF and applicable controls

B5 7 5 1 Spent Nuclear Fuel Transfers from K Basins to the Cold Vacuum Drying Facility

The K Basins sends spent nuclear fuel to the CVDF in accordance with established controls that require a program to maintain compliance between the K Basins TSRs and the TSRs of interfacing facilities through the use of approved and controlled operating specifications and procedures. As discussed in Section 5.7.5.1 of the SNF Project FSAR, SNF shipments from the K Basins must meet the criteria and assumptions identified in Chapter B3.0

B5 7 5 2 Spent Nuclear Fuel Transfers from the Cold Vacuum Drying Facility to the Canister Storage Building

The CVDF sends spent nuclear fuel to the CSB based upon dryness testing (AC 5.17). AC 5.17 requires that proper testing be conducted to establish that the MCO water content is within acceptable levels to ensure that key parameters assumed in the CSB safety analysis are protected. AC 5.16 requires that an acceptable MCO leakage rate be demonstrated before the MCO is shipped to the CSB to ensure CSB leak rate assumptions are protected

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CHAPTER B60

PREVENTION OF INADVERTENT CRITICALITY
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Annex B — Cold Vacuum Drying Facility

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<th>Description</th>
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<tr>
<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
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<tr>
<td>PWC</td>
<td>process water conditioning</td>
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B6 0 PREVENTION OF INADVERTENT CRITICALITY

B6 1 INTRODUCTION

This chapter evaluates the potential for criticality of spent nuclear fuel (SNF) and the fissionable materials in the water drained from the multi-canister overpacks (MCOs) during normal processing and off-normal conditions in the Cold Vacuum Drying Facility (CVDF). It further describes the engineered features and administrative controls that prevent any credible occurrence of a criticality event during handling and processing of the SNF at the CVDF.

Transportation casks containing loaded MCOs are transported on a trailer from the K Basins to the CVDF (HNF-SD-TP-SARP-017). Both the MCO and shipping cask annulus remain flooded with water during transport. The CVDF has four individual processing bays. A tractor-trailer carrying the loaded transportation cask is backed into a bay, the tractor is removed, the bay door is closed, the cask lid is removed, a processing hood is installed, and the water is drained from the MCO. All processing takes place with the cask-MCO on the trailer.

The internal temperature of the MCO is maintained within process limits by circulating water in the cask annulus. The water in the MCO is drained into receiver tanks in the process water conditioning (PWC) system and a helium blanket is maintained in the MCO. Following draining, the MCO is vacuum-dried to remove as much free water as possible. The water in the receiver tanks is filtered, deionized, and stored in the PWC large storage tank (PWC-TK4001). When the tank is full, the water is returned to the K Basins by truck. The design, processing operations, and equipment are described in detail in Chapters B2.0 and B4.0, and a flow diagram is provided in Figure B2-12. The CVDF design aspects that are pertinent to criticality prevention are summarized in Section B6.4, which also includes the administrative controls required to prevent inadvertent criticality.

Many conceivable spent fuel geometrical configurations that could lead to criticality events under both normal and abnormal conditions were postulated. Normal loading of the MCOs and baskets is described in WHC-SD-WM-SAR-062, K Basins Safety Analysis Report. Each MCO may be loaded in one of three normal configurations:

- All fuel baskets
- One scrap basket and four or five fuel baskets
- Two scrap baskets and three or four fuel baskets

Mark IA and Mark IV baskets are not mixed in an MCO. It is permissible to mix Mark IV fuel or scrap with Mark IA fuel or scrap in Mark IA baskets. As described in WHC-SD-WM-SAR-062, the only situations in which Mark IA fuel or scrap would be loaded into Mark IV baskets occur when the 12 long-length Mark IA fuel assemblies in the K West Basin are loaded into a Mark IV fuel basket and when the small amount of Mark IA fuel and scrap in the K East Basin is loaded into Mark IV baskets. Only two types of abnormal conditions for the MCO are considered.
credible at CVDF a misloaded MCO and a flooded MCO with a drained cask–MCO annulus. These normal and abnormal configurations have been analyzed in the criticality safety evaluation report, HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*. The criticality analysis for the PWC system is described in HNF-SD-SNF-CSER-006, *Criticality Safety Evaluation Report for the Cold Vacuum Drying Facility’s Process Water Conditioning System*. Normally, only a small amount of fissile material in suspension is expected to be transferred to the PWC system during the draining and drying process. This material is expected to be removed by a combination of ion exchange and mechanical filtering. The abnormal condition would be for these mechanisms to fail, allowing fissile material to be discharged to the PWC storage tank. A selected set of the bounding analyses applicable to the CVDF is described in Section B6.3. The criticality safety design limits, their bases, and the parameters to be applied for prevention of criticality at the CVDF, are discussed in Section 6.3.3.1 of the SNF Project Final Safety Analysis Report (FSAR).

The double contingency criterion requires at least two unlikely, independent, and concurrent events to occur before a criticality is possible. A contingency is a possible but unlikely change in a condition or control identified as an important barrier in preventing a nuclear criticality accident. The analyses presented in this chapter demonstrate that the double contingency principle is satisfied by showing that the allowed fuel and scrap configurations in the MCO and fissile material buildup in the PWC system will not exceed a $k_{\text{eff}}$ of 0.95 for the violation of any single contingency.

The scope of the criticality analysis presented in this chapter does not include handling of the spent fuel assemblies and the MCOs at the K Basins, transport of the MCOs from the K Basin area to the CVDF, or transport of the MCOs to the Canister Storage Building. This chapter addresses the administrative controls at the K Basins only to the extent those administrative controls, more specifically their failures, could affect criticality safety at the CVDF. The K Basins administrative controls and their failure modes are described in detail in WHC-SD-WM-SAR-062. Criticality information applicable to all SNF Project facilities is provided in Chapter 6.0 of the SNF Project FSAR.


This chapter also addresses implementation of requirements to achieve U.S. Nuclear Regulatory Commission equivalency in the design of the CVDF with respect to criticality prevention. The double contingency requirements in Title 10, *Code of Federal Regulations*, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR 72), Section 72.124, are identical to those delineated in DOE Order 5480.24, *Nuclear Criticality Safety*. 
As discussed in Section B6.3, the $k_{\text{eff}}$ of fuel is determined with full density water, the limiting condition prior to draining. The $k_{\text{eff}}$ of fuel scrap is analyzed for fuel with optimum-sized particles and an optimum water-to-fuel volume ratio prior to draining. The $k_{\text{eff}}$ of the MCO as it is drained is discussed in HNF-SD-SNF-CSER-005. To meet the double contingency criterion the MCOs and the casks are designed to maintain structural integrity and confinement during normal and off-normal conditions, as described in Section B6.3.3.

The $k_{\text{eff}}$ of fissionable material in the process water drained from the MCO is determined by analyzing the optimum fuel-to-moderator ratio of a homogeneous suspension. This optimized suspension was analyzed in the various geometries of the equipment of the PWC system and in a slab on the floor of the PWC room.

Section B6.4 describes the safety features implemented in the design of the spent fuel containers (i.e., fuel baskets and MCOs) and in the vessels and components of the PWC system, to preclude or prevent any criticality event during handling and processing of the MCOs at the CVDF. The administrative controls applied during the operation phase to preclude and prevent criticality are also discussed in Section B6.4.

B6.2 REQUIREMENTS

The requirements that form the basis of criticality protection are identified in Section 6.2 of the SNF Project FSAR.

B6.3 CRITICALITY CONCERNS

B6.3.1 Criticality Hazards

See Section 6.3.1 of the SNF Project FSAR for identification of the fissile material available in SNF Project facilities and loaded into MCOs.

B6.3.2 Multi-Canister Overpack Criticality Analysis Conditions

Each of the four CVDF process bays can contain a single trailer containing a single loaded transportation cask (see Figure B2-8). Neutronic interaction between the loaded cask in each bay is minimized by both the distance between the casks, the bay walls, and by the cask itself. The transportation cask has a shielded central region, a removable lid, and an annulus between the outer wall of the MCO and the inner wall of the transportation cask. The transportation cask is received at the CVDF with the lid installed and the annulus flooded with water. After the trailer is secured in a CVDF process bay, the lid of the cask is removed and a process hood is installed allowing the process equipment to be connected. Two process lines are connected to the...
transportation cask annulus to circulate water through the annulus region for temperature control while the water inside the MCO is pumped out through the process suction line connected to the long axial process tube in the MCO. The MCOs are sealed after drying for transport to the Canister Storage Building.

Under normal conditions, each type of spent fuel is contained in the baskets designated for that type of fuel (see Figures 6-1 and 6-2 in the SNF Project FSAR), and the baskets in turn are stacked in an MCO (see Figure 6-3 in the SNF Project FSAR). The Mark IA basket holds a maximum of 48 Mark IA fuel assemblies and has a central, 6.625-in. outer diameter, stainless steel center post specifically designed for criticality control. The Mark IA fuel scrap basket has a similar center post. This center post reduces the amount of fuel or scrap the MCO basket can contain to a critically favorable geometry in the MCO and has a hole 1.75 in. in diameter in the center for the long axial process tube. The Mark IV fuel basket holds a maximum of 54 fuel assemblies. Neither the fuel basket nor the Mark IV scrap basket have the centrally located pipe for criticality control but do have a 2.6-in. outside diameter, centrally located tube for installation of the long axial process tube used for draining and vacuum drying the MCO.

The MCO has a shield plug installed and is always inside a transportation cask while at the CVDF, the transportation cask lid is removed at the CVDF. For the purposes of criticality analysis, the configuration of a loaded MCO with the largest \( k_{\text{eff}} \) was determined. For both a Mark IA and Mark IV MCO, this configuration includes two scrap baskets, one at the top and the other at the bottom, with partially loaded fuel baskets. Each partially loaded Mark IA fuel basket contains 47 fuel assemblies with an empty location in the middle row of the fuel assemblies, and each partially loaded Mark IV fuel basket contains 53 fuel assemblies with an inner element only loaded in the outer fuel assembly row (Figures 6-4 and 6-5 of the SNF Project FSAR). Each processing bay may contain only one transportation cask, but all processing bays may contain a transportation cask at the same time.

A complete description of the codes used to perform the criticality analyses, their modeling assumptions, and their validations is provided in HNF-SD-SNF-CSER-005. The effect on \( k_{\text{eff}} \) of changes in moisture content, fuel length selections, corrosion, and sludge deposition have been investigated in HNF-SD-SNF-CSER-005 by performing calculations over the range of possible variations of the individual parameters. These calculations show that \( k_{\text{eff}} \) typically increases as the fuel length approaches the basket diameter. The \( k_{\text{eff}} \) decreases as fuel length gets greater than the basket diameter in the case of Mark IV fuel. Corrosion of the fuel reduces \( k_{\text{eff}} \) because the uranium oxide that is produced is neutronically less reactive than the metal fuel. The longest length assemblies for the Mark IV fuel (26.1 in. long) and the 20.9-in. long elements for the Mark IA fuel were chosen to represent "normal conditions" to compensate for these uncertainties. Only 12 Mark IA fuel assemblies are left in the K West Basin that are longer than 20.9 in. There is also a small amount of Mark IA fuel and scrap in the K East Basin. Normally, Mark IA material is not loaded into Mark IV baskets. However, HNF-SD-SNF-CSER-010, Criticality Safety Evaluation Report for the K Basin Fuel Retrieval Subproject, allows the loading of Mark IA fuel and scrap into Mark IV baskets under two specific conditions. In the K West Basin, the twelve 26.1-in. long Mark IA assemblies may be loaded into a Mark IV fuel basket containing...
42 Mark IV fuel assemblies. The small amount of Mark IA fuel and scrap in the K East Basin also may be loaded into Mark IV fuel or scrap baskets. A maximum of seven Mark IA fuel assemblies may be loaded into a Mark IV fuel basket containing 47 Mark IV fuel assemblies. Mark IA scrap may be loaded into a Mark IV scrap basket and is limited to 105 kg. The total mass in the basket may not exceed 980 kg (Kessler and Peck 1999).

The potential exists for significant amounts of fuel material to be entrained in the MCO water, and the PWC system may contain sludge or particulate from an MCO. Therefore, after each MCO is drained, the lines are flushed and drained to the receiver tank and the water is processed and monitored before being sent to the storage tank. Water in the PWC system is drained to the storage tank before processing the next MCO. The criticality analysis documented in HNF-SD-SNF-CSER-006 assumed 15 kg of material is transferred to the PWC system. According to the test results using cerium oxide, 6% to 7% of the particulate will be transferred from the MCO to the PWC system. For conservatism, the safety basis assumes a maximum of 1.5 kg of 1.25 wt% $^{235}$U as UO$_2$ enters the system for each MCO processed (SNF-2770).

B6 3.3 Multi-Canister Overpack Criticality Analysis Results

This section presents a summary of the results of the analyses performed to evaluate the potential for criticality in the MCO under normal and off-normal conditions. The specifics are documented in HNF-SD-SNF-CSER-005. The scrap basket designs have been modified to include a small region around the center post to contain small pieces of fuel that make it through the cleaning process. Since these particles are approximately the same size as optimized scrap and the criticality analyses assumed the scrap baskets were loaded with optimized scrap, no additional analyses have been performed using the new design.

This section also provides the bases for establishing the criticality safety design limits, the requirements for engineered design features, and the administrative controls for prevention of inadvertent criticality. The double contingency requirements were implemented to ensure that more than one unlikely, independent, and concurrent event has to occur to challenge criticality safety during handling and processing of the spent fuel contained in the MCOs.

The following sections summarize the criticality scenario conditions that need to be addressed for the CVDF. Sections B6 3.3.1 and B6 3.3.2 address all credible MCO scenarios. Drop scenarios are not considered credible at the CVDF because the MCO is not lifted, and the cask will not spill or fall from the transporter during a design basis earthquake (HNF-2179).

B6 3.3.1 Normal Conditions — Multi-Canister Overpack Handling

This section summarizes the results of the analyses for normal operating conditions. At the CVDF, the MCO is processed in seven stages:

- Bypass
- Heatup

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Section B2 5 1 presents an overview of the drying process. The seven stages of the drying process correlate to the switch positions on the safety-class instrumentation and control system panels. Table B6-1 lists the condition of the cask-MCO and the maximum calculated k\text{eff} for partially loaded MCOs at each stage. These calculations do not include the long-length Mark IA assemblies loaded into Mark IV baskets.

**B6 3 3 1 1 Multi-Canister Overpacks during Draining Process** The maximum k\text{eff} of the MCO exists before draining. As the MCO is drained, k\text{eff} initially decreases then increases very slightly as neutrons are reflected from drained regions to flooded regions. As the water is drained from the lower fuel baskets, k\text{eff} decreases. The peak k\text{eff} during the drain does not exceed the k\text{eff} of a flooded MCO. These results are shown in Tables B6-2 and B6-3 and Figure B6-1.

**B6 3 3 1 2 Drained Loadings** Cases cvd1 8 and cvd4 8 in Table B6-1 represent the final drained state of a transportation cask containing a fully loaded MCO at the CVDF. These dry cases assume 3 kg of water with a water density of 0.005 g/cm^3 as residual water left in the MCO after drying.

**B6 3 3 2 Abnormal Conditions for Multi-Canister Overpacks at the Cold Vacuum Drying Facility** Two bounding abnormal conditions at the CVDF were analyzed: draining the cask annulus while the MCO is still flooded, and receipt of a misloaded MCO. Three types of misloaded MCOs were analyzed: an MCO containing two adjacent scrap baskets, an MCO containing an extra scrap basket, or a Mark IV MCO containing a single basket misloaded with one canister of Mark IA fuel or scrap. As discussed in HNF-SD-SNF-CSER-005, the credible misloaded basket scenarios are for one Mark IA canister containing 14 Mark IA fuel assemblies to be loaded into a Mark IV fuel basket, or for the mass equivalent of 14 Mark IA assemblies, 233 kg, to be loaded into a Mark IV scrap basket. A receipt inspection will be performed on the MCO at the CVDF. Loading of the MCOs is done in the K Basins and the technical safety requirements for loading are described in Chapter 50 of the K Basins safety analysis report (WHC-SD-WM-SAR-062). The quality assurance documentation for MCO loading will be verified to ensure loading corresponds to the approved configurations for CVDF. The analyses of these abnormal conditions are summarized below.
Table B6-1  Criticality Analysis Results for Cold Vacuum Drying Facility Normal Operating Conditions

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<th>Cask–MCO processing sequence (see Chapter B2 0 for descriptions)</th>
<th>Loaded MCO condition</th>
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<th>$\sigma_{calc}$</th>
<th>$k_{eff}$(^b)</th>
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<td>Drain Purge and flush Dry Proof Pressure test</td>
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<td>Mark IV cvd4 8</td>
<td>0.31643</td>
<td>0.00051</td>
<td>0.32743</td>
</tr>
</tbody>
</table>

Note: The MCO was modeled with partially loaded baskets to maximize the $k_{eff}$.


\[^{a}\] $k_{eff} = k_{calc} + 0.0004 + [(0.01)^2 + 1.645^2 (\sigma^2 + 0.002083^2)]^{0.5}$ as defined in Section 6 3 3 of the SNF Project FSAR. This value should be less than or equal to the criticality prevention criterion of 0.95.

\[^{b}\] Regions internal to the MCO have a water density of 0.0051 g/cm\(^3\).

\[^{c}\] The cask annulus has a water density of 0.0012 g/cm\(^3\).

CSB = Canister Storage Building
CSER = criticality safety evaluation report
MCO = multi-canister overpack
Table B6-2 Criticality Analysis Results for Mark IA Multi-Canister Overpack during Draining

<table>
<thead>
<tr>
<th>Case</th>
<th>Configuration analyzed</th>
<th>$k_{calc}$</th>
<th>$\sigma_c$</th>
<th>$k_{eff}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>mk1r 2</td>
<td>Base case with the highest $k_{eff}$</td>
<td>0.89017</td>
<td>0.00096</td>
<td>0.90126</td>
</tr>
<tr>
<td>cvd1 0</td>
<td>Top scrap basket drained</td>
<td>0.88691</td>
<td>0.00099</td>
<td>0.89801</td>
</tr>
<tr>
<td>cvd1 1</td>
<td>Top scrap basket and tier 5 fuel basket drained</td>
<td>0.88788</td>
<td>0.00103</td>
<td>0.89899</td>
</tr>
<tr>
<td>cvd1 2</td>
<td>Top scrap basket and tier 4 and 5 fuel baskets drained</td>
<td>0.88832</td>
<td>0.00095</td>
<td>0.89941</td>
</tr>
<tr>
<td>cvd1 3</td>
<td>Top scrap basket and tier 3, 4 and 5 fuel baskets drained</td>
<td>0.88618</td>
<td>0.00101</td>
<td>0.89728</td>
</tr>
<tr>
<td>cvd1 4</td>
<td>Top scrap basket and all fuel baskets drained</td>
<td>0.88724</td>
<td>0.00107</td>
<td>0.89836</td>
</tr>
<tr>
<td>cvd1 9</td>
<td>Bottom scrap basket half full</td>
<td>0.80847</td>
<td>0.00099</td>
<td>0.81957</td>
</tr>
<tr>
<td>cvd1 6</td>
<td>Drained MCO with a flooded annulus</td>
<td>0.34606</td>
<td>0.00061</td>
<td>0.35708</td>
</tr>
<tr>
<td>cvd1 8</td>
<td>Drained MCO with a drained annulus</td>
<td>0.27991</td>
<td>0.00047</td>
<td>0.29091</td>
</tr>
</tbody>
</table>

*a $k_{calc}$ is based on the partially loaded MCO
*b $k_{eff} = k_{calc} + 0.0004 + ((0.01)^2 + (1.645)^2(\sigma_c^2 + 0.002083))$ as defined in Section 6.3.3 of the SNF Project FSAR
This value should be less than or equal to 0.95

MCO = multi canister overpack

Table B6-3 Criticality Analysis Results for Mark IV Multi-Canister Overpack during Draining

<table>
<thead>
<tr>
<th>Case</th>
<th>Configuration analyzed</th>
<th>$k_{calc}$</th>
<th>$\sigma_c$</th>
<th>$k_{eff}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>mk4r 4</td>
<td>Base case with the highest $k_{eff}$</td>
<td>0.91540</td>
<td>0.00084</td>
<td>0.92646</td>
</tr>
<tr>
<td>cvd4 0</td>
<td>Top scrap basket drained</td>
<td>0.91231</td>
<td>0.00090</td>
<td>0.92338</td>
</tr>
<tr>
<td>cvd4 1</td>
<td>Top scrap basket and tier 4 fuel basket drained</td>
<td>0.91372</td>
<td>0.00087</td>
<td>0.92479</td>
</tr>
<tr>
<td>cvd4 2</td>
<td>Top scrap basket and tier 3 and 4 fuel baskets drained</td>
<td>0.91351</td>
<td>0.00091</td>
<td>0.92459</td>
</tr>
<tr>
<td>cvd4 3</td>
<td>Top scrap basket and all fuel baskets drained</td>
<td>0.91204</td>
<td>0.00083</td>
<td>0.92310</td>
</tr>
<tr>
<td>cvd4 9</td>
<td>Bottom scrap basket half full</td>
<td>0.85086</td>
<td>0.00086</td>
<td>0.86193</td>
</tr>
<tr>
<td>cvd4 6</td>
<td>Drained MCO with a flooded annulus</td>
<td>0.36980</td>
<td>0.00062</td>
<td>0.38082</td>
</tr>
<tr>
<td>cvd4 8</td>
<td>Drained MCO with a drained annulus</td>
<td>0.31643</td>
<td>0.00051</td>
<td>0.32743</td>
</tr>
</tbody>
</table>

*a $k_{calc}$ is based on the partially loaded MCO
*b $k_{eff} = k_{calc} + 0.0004 + ((0.01)^2 + (1.645)^2(\sigma_c^2 + 0.002083))$ as defined in Section 6.3.3 of the SNF Project FSAR
This value should be less than or equal to 0.95

MCO = multi canister overpack
6 3 3 2 1 Draining the Annulus of a Flooded Multi-Canister Overpack. If the cask annulus is drained before the MCO is dried, reflection of neutrons by the cask back into the MCO increases, and \( k_{\text{eff}} \) will increase but remain below 0.95. This scenario is shown in Table B6-4. Included in this table are the results for the long-length Mark IA assemblies that will be loaded into Mark IV fuel baskets containing Mark IV fuel. Eighteen Mark IA assemblies were modeled, which bounds the actual loading of 12 Mark IA assemblies. The assemblies were modeled in all rows of the fuel basket to bound all possible loadings. The fuel baskets were partially loaded giving the largest \( k_{\text{eff}} \) (see Section 6 3 2 of the SNF Project FSAR) and the largest \( k_{\text{eff}} \) was calculated for the case with the Mark IA assemblies in the center of the bottom fuel basket.

Table B6-4 Calculation Results for Flooded Multi-Canister Overpack Received at the Cold Vacuum Drying Facility in Shipping Cask with Drained Annulus

<table>
<thead>
<tr>
<th>Case</th>
<th>MCO configuration</th>
<th>( k_{\text{calc}} )</th>
<th>( \sigma_\text{c} )</th>
<th>( k_{\text{eff}} ^a )</th>
</tr>
</thead>
<tbody>
<tr>
<td>cvd1 7</td>
<td>2 scrap baskets and 4 partially loaded fuel baskets in a Mark IA MCO</td>
<td>0.90154</td>
<td>0.00096</td>
<td>0.91263</td>
</tr>
<tr>
<td>cvd4 7</td>
<td>2 scrap baskets and 3 partially loaded fuel baskets in a Mark IV MCO</td>
<td>0.92152</td>
<td>0.00088</td>
<td>0.93259</td>
</tr>
<tr>
<td>18rc 15</td>
<td>18 long-length Mark IA fuel assemblies(^c) in the center of the lower Mark IV fuel basket next to the lower scrap basket</td>
<td>0.92453</td>
<td>0.00087</td>
<td>0.93560</td>
</tr>
<tr>
<td>18rp 2</td>
<td>18 long-length Mark IA fuel assemblies(^b) in the outer row of the center Mark IV fuel basket</td>
<td>0.92331</td>
<td>0.00091</td>
<td>0.93439</td>
</tr>
<tr>
<td>18rm 15</td>
<td>18 Mark IA fuel assemblies in middle row of lower Mark IV fuel basket next to the lower scrap basket</td>
<td>0.92250</td>
<td>0.00085</td>
<td>0.93356</td>
</tr>
</tbody>
</table>

\( k_{\text{calc}} \) is based on the partially loaded MCO

\( \Delta k_{\text{eff}} = k_{\text{calc}} + 0.0004 + ((0.01)^2 + (1.645)^2(\sigma_\text{c}^2 + 0.002083^2)) \)\(^a\). This value should be less than or equal to 0.95.

\(^c\)Eighteen long length Mark IA assemblies were used in the analysis to be conservative

\(^b\)MCO = multi canister overpack

6 3 3 2 2 Misloaded Multi-Canister Overpacks or Multi-Canister Overpack Baskets

Baskets are loaded into the MCOs by the MCO cask loading system at the K Basins. There are no engineered features that prevent loading scrap baskets adjacent to each other or loading extra scrap baskets in an MCO. The system does sense the basket stack height, which prevents loading Mark IA and Mark IV baskets into the same MCO. Detailed analyses for misloaded MCOs are discussed in HNF-SD-SNF-CSER-005, Sections 5 2 3 and 5 2 4, using the reference case with the highest \( k_{\text{eff}} \) for partially loaded fuel baskets. For all credible scenarios involving misloaded
MCOs, $k_{\text{eff}}$ stays well below 0.95. The most limiting scenario is loading two Mark IV scrap baskets adjacent to each other, which has a $k_{\text{eff}}$ of approximately 0.934. There are also no engineered features that prevent loading Mark IA fuel or scrap into a Mark IV basket. The effect of loading a canister of Mark IA fuel or scrap into a Mark IV basket has been evaluated and shown not to exceed a $k_{\text{eff}}$ of 0.926 (HNF-SD-SNF-CSER-005). These results are listed in Table B6-5 for the misloaded cases with the highest $k_{\text{eff}}$.

**Table B6-5** Criticality Analysis Results for Misloaded Multi-Canister Overpacks

<table>
<thead>
<tr>
<th>Case</th>
<th>MCO misloaded</th>
<th>$k_{\text{calc}}$</th>
<th>$\sigma_e$</th>
<th>$k_{\text{eff}}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>mk4rl 4</td>
<td>14 Mark IA fuel assemblies in the outer row of the tier 2 Mark IV fuel basket</td>
<td>0.91357</td>
<td>0.00092</td>
<td>0.92465</td>
</tr>
<tr>
<td>mk4rc 14</td>
<td>The mass of 14 Mark IA outer elements (155 kg at 25 wt% enriched scrap) in the center of the upper Mark IV scrap basket</td>
<td>0.91485</td>
<td>0.00097</td>
<td>0.92594</td>
</tr>
<tr>
<td>mk1rs 8</td>
<td>Three scrap baskets loaded in the bottom three tiers of a Mark IA MCO</td>
<td>0.89359</td>
<td>0.00098</td>
<td>0.90468</td>
</tr>
<tr>
<td>mk4rs 1c</td>
<td>Scrap baskets in tiers 1, 2, and 5 of a Mark IV MCO</td>
<td>0.92256</td>
<td>0.00082</td>
<td>0.93362</td>
</tr>
</tbody>
</table>

- $k_{\text{calc}}$ is based on the partially loaded MCO
- $k_{\text{eff}} = k_{\text{calc}} + 0.0004 + ((0.01)^2 + (1.645)^2)(\sigma_e^2 + 0.002083^2)$ as defined in Section 6.3.3 of the SNF Project FSAR. This value should be less than or equal to 0.95.
- The MCO drop model containing five scrap baskets bounds all misloading for the Mark IV MCO.

**B6 3.4 Process Water Conditioning System Criticality Analysis Conditions**

The liquid in the MCO is drawn with a water jet ejector into the PWC receiver tank loop, which cycles water from the receiver tanks, through the water jet ejector, and back into the receiver tanks. The liquid is then cycled through the ion exchange modules and back into the receiver tank. When the liquid is sufficiently cleaned by the ion exchange modules, it is pumped from the receiver tank, through the ion exchange modules, through a filter, and into the storage tank.

On the average, it is expected that approximately 66 g of fissionable material oxides will be generated in an MCO from the time the fuel is cleaned in the K Basins until the water in the MCO is drained (HNF-SD-SNF-TI-015). In the bounding case, it is calculated that approximately 15 kg of fissionable material oxide could be generated (SNF-2770, Chapter 2.0). Because the combination of helium pressure and water jet ejector suction is low, draining the MCO takes approximately 30 minutes, and the only material transferred to the PWC system will be small.
particles in suspension. The small particles contained in the center region of the scrap baskets will not be removed from the MCO during the draining process, and no additional analyses were performed after the scrap baskets were redesigned to isolate a small region around the center post to contain fuel “fines.” Modeling the scrap basket as filled with optimized scrap bounds calculations that model the fines center region of the basket. As discussed in Chapter B3.0, approximately 10% or 1.5 kg, of the oxide generated after cleaning will be transported by the MCO process water into the PWC system during draining (SNF-2770). If each MCO released 1.5 kg of UO₂ into the PWC system, the total amount of UO₂ would be 660 kg. This is approximately 75% of the minimum spherical 0.95 mass of 875 kg. Clearly, a criticality in the PWC system, any of its components, or on the floor of the PWC room is impossible. However, the criticality analysis assumed that a maximum amount of 15 kg was transported to the PWC system (HNF-SD-SNF-CSER-006).

B6 3.5 Process Water Conditioning System Criticality Analysis Results

The components of the PWC system have been evaluated for criticality safety in HNF-SD-SNF-CSER-006. Dimensional limits have been imposed to ensure geometrically favorable components. All geometrically favorable piping, pumps, receiver tanks, and filters have a maximum inner diameter of 23.5 in. Interaction between components has been evaluated (HNF-SD-SNF-CSER-006), and even if the receiver tanks, filter, and piping have no spacing between them and the PWC room is flooded, $k_{\text{eff}}$ does not exceed 0.62.

The ion exchange modules are designed in accordance with design specification HNF-3904, *Procurement Specification for Ion Exchange Modules*, and have been shown to be geometrically favorable (WHC-SD-NR-CSER-011). Their maximum height is 42.25 in, and each column has an outer diameter of 16 in, with a 0.25-in wall thickness. The columns are housed in concrete and have been shown not to interact with other equipment (WHC-SD-NR-CSER-011). To exceed a $k_{\text{eff}}$ of 0.95, the ion exchange module would have to collect a minimum of 13 kg of PuO₂ with no UO₂ (WHC-SD-NR-CSER-011). The corrosion products in the MCO will contain both UO₂ and PuO₂. Typical uranium-to-plutonium ratios for the sludge in the K East Basin, where uncapped canisters are stored is greater than 400 to 1 (HNF-SD-SNF-TI-009). At a 400 to 1 ratio, the safe mass for an optimally moderated, optimally reflected hemisphere is 77.88 kg of PuO₂ and 31,022 kg of UO₂ (WHC-SD-NR-CSER-014). Based on 1.5 kg of oxide per MCO, a total of 660 kg of particulate could be carried over into the PWC system. Thus, if the ion exchange module collected all the particulate from each MCO, it would contain 1.6 kg of PuO₂ and 658 kg of UO₂, well below the safe mass for the UO₂ - PuO₂ - H₂O system.

The limit on the total mass in both the storage tank and the system is 875 kg of 1 25 wt% enriched UO₂ (HNF-SD-SNF-CSER-006).

B6 3.5.1 Normal Process The receiver tanks have a capacity of 300 gal (1,140 L), with an initial 23 gal (120 L) of water used to prime the water jet ejector. Flow from the receiver tanks circulates from the tanks, through the pumps, through the water jet ejector, and returns to the
The water jet ejector draws approximately 165 gal (625 L) of water from the MCO into the receiver tank loop. The emptying of the MCO water occurs over approximately 30 minutes. Once the water and fissionable material particulate from the MCO are in the receiver tank, the drain lines from the MCO to the receiver tanks are flushed with clean water to remove any buildup of contaminants. When this step is completed, the receiver tanks contain about 300 gal of water.

After the MCO is drained, the process water is pumped from the receiver tanks, through the ion exchange modules, and back into the receiver tanks. During the process, samples will be taken upstream and downstream from the ion exchange modules to determine the removal efficiency and the ion exchange module loading. The water will also be sampled after processing before being pumped to the storage tank. Personnel will transport these samples to a laboratory for analysis. The samples will be analyzed for fissionable material content. There are no criticality controls for transporting these samples (HNF-SD-SNF-CSER-006).

The water in the storage tank will be transferred to a tanker truck when the tank is full (15 to 20 MCOs have been processed) and taken to the K Basins. Because this truck may be used to transfer liquids at other facilities, limits are placed on the transfer by HNF-SD-SNF-CSER-006. If the truck has not been used to transfer water from sources other than CVDF, there are no limits on the transfer from CVDF. However, if the truck has been used for other transfers, the following limits apply before water is transferred from the PWC storage tank to the truck:

- The truck tank must be verified to have been cleaned and flushed following any transfer from a facility other than the CVDF.
- The water left in the truck as a "heel" must be sampled and the following verified:
  - Particle size is less than 500 μm
  - The amount of fissile material is less than 200 kg

**B6 3.5.2 Abnormal Conditions** The following abnormal conditions in the PWC system were evaluated from a criticality standpoint:

- Ion exchange module failure (release of particulate or resin into the system)
- Chronic or acute particulate filter failure
- Seismic event causing spill of PWC system contents to the floor
- Unsafe amount of fissionable material accumulated in the storage tank

The criticality considerations for these accidents are addressed with a combination of geometrically favorable components and controls on the amount of material allowed to build up in the storage tank for defense in depth. These considerations are discussed in Section B6 4.2.
B6 4 CRITICALITY CONTROLS

B6 4 1 Multi-Canister Overpack Criticality Controls

B6 4 1 1 Multi-Canister Overpack Engineering Controls  The MCO and its internals, including fuel baskets, are designed to maintain their structural integrity and perform their safety function under credible accident conditions, such as a design basis earthquake or a design basis accident impact during an MCO drop accident. The design details for the MCO and its internals are summarized in HNF-SD-SNF-SARR-005, Multi-Canister Overpack Topical Report.

Various components of both the baskets and MCOs are required to be safety class. These include the MCO shell and shield plug assembly, including the internal filter guard and the center post and baseplate of the Mark IA fuel and scrap baskets. None of the components of the Mark IV fuel and scrap baskets have been identified in Chapter B4 O as being safety class.

B6 4 1 2 Multi-Canister Overpack Administrative Controls  Basket loading and MCO loading will be performed in the K Basins. Criticality evaluations have been performed that specify the limits for these evolutions. Fuel basket loadings are limited by basket design to 48 Mark IA assemblies and 54 Mark IV assemblies. Scrap basket limits are 575 kg for Mark IA scrap and 980 kg for Mark IV scrap (HNF-SD-SNF-CSER-010). Additional analyses were performed to show it is acceptable to load the twelve 26.1-in.-long Mark IA fuel assemblies into a Mark IV fuel basket. Table B6-4 shows the results of these analyses for the contingency of a drained cask-MCO annulus at the CVDF. These results bound the expected loading (HNF-SD-SNF-CSER-005).

Loading the Mark IA material stored in the K East Basin into Mark IV baskets has also been evaluated and shown to be acceptable (HNF-SD-SNF-CSER-010). Table B6-5 shows the results of the credible misloading of Mark IA material into Mark IV fuel or scrap baskets. These results bound the loading of the small amount of Mark IA material in the K East Basin into Mark IV baskets. The MCO handling in the CVDF will not affect the geometric configuration of the fuel or baskets (HNF-SD-SNF-CSER-005). There are no operations involving lifting the MCO off of the transporter, and the transporter is designed not to fail during a seismic event (HNF-2179).

B6 4 1 3 Application of Double Contingency Principle in the Multi-Canister Overpack  Abnormal events for the MCO at the CVDF are identified in Section B6 3 3 2. Correct basket and MCO loading will be verified at the K Basins before the MCO is shipped to the CVDF and will be verified again by reviewing the loading documentation upon receipt at CVDF before the MCO is connected to the PWC system. As shown in Table B6-5, the maximum $k_{eff}$ for a credible misloaded MCO is well below the 0.95 limit. The only other abnormal event is one in which the cask annulus is left drained while the MCO remains flooded. Even should this happen, the $k_{eff}$ will stay well below the 0.95 limit as shown in Table B6-4. Thus, a single event will not exceed the safety limit.
B6 4 2  Process Water Conditioning System Controls

The following sections describe PWC system requirements that provide defense in depth.

B6 4 2.1  Process Water Conditioning System Engineering Controls For defense in depth, the receiver tanks, pumps and piping, ion exchange modules, and filter housing of the PWC system have been designed to be geometrically favorable (HNF-SD-SNF-CSER-006, HNF-SD-SNF-CSER-009).

B6 4 2.2  Process Water Conditioning System Administrative Controls Three defense-in-depth requirements have been established to monitor the potential for criticality in the PWC system.

- Samples of the water drained from each MCO will be taken upstream and downstream of the ion exchange module and on the inlet to the storage tank to determine concentrations and accumulations of fissionable materials and effectiveness of the PWC equipment.
- Radiation levels in the PWC room will be monitored by area radiation monitors.
- The storage tank will be flushed to prevent fissionable material buildup. The requirements for flushing will be specified in the criticality safety specifications.

B6 4 2.3  Application of Double Contingency Principle in the Process Water Conditioning System Section B6 3 5 2 identifies abnormal events for the PWC system, which include component failure. Each event is analyzed for criticality safety using the bounding limit of the fissionable material, 15 kg. For each event, the component is assumed to be at its process limit before the event occurs. Criticality safety is maintained because no single event can cause a criticality. Additionally, if the amount of material entering the system from a single MCO was at the bounding value of 15 kg, radiation levels in the PWC room would reach unacceptable levels (HNF-2850). This would occur long before any criticality limits are approached because the water contains radioactive fission products not considered in the criticality analysis. Using a maximum expected value of 1.5 kg per MCO and assuming an enrichment of 1.25 wt% $^{235}\text{U}$ shows that there is not enough material removed from all 440 MCOs to produce a minimum critical mass in any of the PWC components (i.e., 660 kg total removed where 875 kg is the minimum fuel mass that will exceed the $k_{\text{eff}}$ of 0.95). For spills to the PWC room floor to a depth of 10.5 in at optimum conditions of moderation, enrichment, and reflection (HNF-SD-SNF-CSER-009), the amount of fissionable material required for $k_{\text{eff}}$ to exceed 0.95, 43,000 kg, is greater than the maximum amount of material available in each of the 440 MCOs, making this event incredible.
B6 4 3 Design Basis Earthquake

A design basis earthquake has a very low probability of occurrence and little impact on the MCO and PWC system criticality (see Chapters B2 0, B3 0, B4 0, and the preceding paragraphs). The design basis earthquake is defined in Chapter B1 0. A design basis earthquake will cause slight structural damage to the CVDF building, but the transportation cask geometry, which is required for criticality safety, will remain functional (see Chapter B2 0).

B6 5 CRITICALITY PROTECTION PROGRAM

An overview of the organizational structure and interfaces and the technical and administrative practices of the criticality protection policy and programs that are developed for SNF Project operations is provided in Section 6 5 of the SNF Project FSAR. There are no unique requirements for the CVDF.

B6 6 CRITICALITY INSTRUMENTATION

A criticality alarm system or criticality detection system is not required for the CVDF in either the process bays or the PWC room. ANSI/ANS-8 3-1997, Criticality Accident Alarm System, and DOE Order 5480 24 state that neither a criticality alarm nor criticality detection system is required where the probability of a criticality accident is determined to be less than $1 \times 10^{-6}$ per year. Interpretive guidance on the probability determination (Holten 1993) states that “the use of $10^{-6}$ does not necessarily mean that a PRA [probabilistic risk assessment] has to be performed. Reasonable grounds shall be presented on the basis of commonly accepted engineering judgement.”

Section 5 2 of HNF-SD-SNF-CSER-006 contains a detailed evaluation of the PWC room and its components. The only component in the PWC system without a critically favorable design is the 5,000-gal storage tank. Only two scenarios could lead to a significant buildup of fissionable material in the storage tank:

- Both the ion exchange module and filter fail undetected while processing several MCOs.
- The transfer of liquid from the storage tank to the transfer truck does not contain any fissionable material and this accumulates in the bottom of the storage tank over time.

In addition, area radiation monitors in the PWC room must fail or be ignored by the operators, and the health physics technicians must fail to detect or report high radiation readings during routine surveys. Based on a maximum of 1.5 kg of fissionable material released to the PWC system per MCO (SNF-2770), only 660 kg of fissionable material could build up in the PWC system for the entire campaign of 440 MCOs. This is less than the minimum 0.95 mass of 875 kg.
The process bays have also been evaluated, and the probability of a criticality accident has been shown to be less than $10^{-4}$ per year based on engineering judgement. Each process bay can only contain a single cask-MCO on a trailer. The highest $k_{\text{eff}}$ that results from the worst-case accident, a drop of a flooded MCO in which all the fuel rubbilizes to optimized scrap, is less than 0.94 (HNF-SD-SNF-CSER-005). An evaluation has also been performed for the process bay sumps, which normally collect uncontaminated rain water and snow melt that drops from the trailer. The sumps are small, 5 ft by 5.5 ft by 2 ft. Based on the safe slab height of 10.5 in (HNF-SD-SNF-CSER-009), the sumps would have to contain more than 2,300 kg of 12.5 wt% enriched UO$_2$ for a criticality to be possible. This would require all the material from more than all 440 MCOs to be drained to the sump rather than to the PWC system. It further requires that the sumps not be cleaned out and that high radiation levels in the process bay be ignored, clearly an incredible series of events.

The liquid in the sumps is either pumped into drums for disposal or gravity drained to a retention basin outside the CVDF. Pumping the liquid containing the maximum expected amount of material from an MCO into a drum would produce high radiation levels. This would have to be done repeatedly, and the high radiation levels ignored, in order for a criticality to be possible (HNF-SD-SNF-CSER-006). For a criticality to be possible in the retention basin, over 50 metric tons of fissionable material would have to be drained from the sumps. This is more material than can possibly be extracted from 440 MCOs (HNF-SD-SNF-CSER-006). These events are clearly incredible. Therefore, neither a criticality alarm nor criticality detection system is required.

B6.7 U.S. Nuclear Regulatory Commission Equivalency

The U.S. Nuclear Regulatory Commission equivalency in HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements, is implemented in the design of the CVDF for criticality prevention considerations by using a $k_{\text{eff}}$ limit of 0.95. This is the only requirement from HNF-SD-SNF-DB-003 related to criticality for the CVDF.

B6.8 References


HFN-3553 REV 0
Annex B — Cold Vacuum Drying Facility


Figure B6-1  K-Effective During Draining of the Multi-Canister Overpack

Note  The error bars indicate ± two standard deviations from the calculated values
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CHAPTER B7 0

RADIATION PROTECTION
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<th>Term</th>
<th>Definition</th>
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<td>ALARA</td>
<td>as low as reasonably achievable</td>
</tr>
<tr>
<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>HEPA</td>
<td>high-efficiency particulate air (filter)</td>
</tr>
<tr>
<td>HVAC</td>
<td>heating, ventilation, and air conditioning</td>
</tr>
<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
</tr>
<tr>
<td>NRC</td>
<td>U S Nuclear Regulatory Commission</td>
</tr>
<tr>
<td>PWC</td>
<td>process water conditioning</td>
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<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
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B7 1 INTRODUCTION

The essential features of radiation protection programs that provide for radiation exposure control, radiological monitoring, and radiological protection instrumentation at all the Spent Nuclear Fuel (SNF) Project facilities are addressed in Chapter 7 of the SNF Project Final Safety Analysis Report (FSAR). Additional features of radiation protection specific to the Cold Vacuum Drying Facility (CVDF) are addressed in this Chapter B7. The CVDF’s primary function is to dry the spent fuel packed within the multi-canister overpack (MCO) to make it safe for transport, and for staging and interim storage at the Canister Storage Building. Water must be controlled to limit pressure in the MCO and the buildup of hydrogen during transportation, staging, and interim storage. A shipping cask–MCO is loaded with fuel underwater at the K Basins and transported still filled with water to the CVDF. The cask–MCO is transported to the CVDF by a tractor and trailer, and processing at the CVDF occurs with the cask–MCO remaining on the trailer. The CVDF process drains the basin water from the MCO and performs vacuum drying to remove as much additional water as possible from the MCO. At the end of the vacuum drying process, each MCO is tested for pressurization rate to ensure safe MCO conditions for the Canister Storage Building.

The major risk of radiation exposure occurs when the cask lid is removed to allow connection and removal of the seal ring and process hood and connecting and disconnecting the process connectors. With the cask lid off, the top of the MCO shield plug is exposed. Radiation fields within the radius of the cask above the MCO are expected to be less than 5 mrem/h (less than 0.05 mSv/h) for the hands and arms used to perform the process line connection and disconnection operations. Dosimetry records for personnel working near the uncovered MCO will be maintained initially for both beta/gamma and neutron exposures. Based on the review of the dosimetry results from the validation program, monitoring of neutron doses may be discontinued. During normal drying operations, processing is controlled remotely from the control room, and personnel are required to be in the process bay only for routine operational surveillances. Exposures to airborne radioactive material are expected to be minimal.

B7 2 REQUIREMENTS

The requirements that form the basis for the radiation protection program are identified in Section 7 2 of the SNF Project FSAR.

B7 3 RADIATION PROTECTION PROGRAM AND ORGANIZATION

The SNF Project radiation protection program and its organization, including safety management policies and philosophies, are described in Section 7 3 of the SNF Project FSAR.
B7.4 ALARA POLICY AND PROGRAM

See Section 7.4 of the SNF Project FSAR for a discussion of the SNF Project ALARA (as low as reasonably achievable) policy and program. The SNF Project policy regarding ALARA principles during CVDF design and construction is described in HNF-3552, *Spent Nuclear Fuel Project Execution Plan*, HNF-SD-SPN-DB-003, *Spent Nuclear Fuel Project Path Forward Additional NRC Requirements*, requires that the SNF Project incorporate U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 8.8, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable*. The NRC Regulatory Guide 8.8 requirements were appropriately implemented in the design during the design evaluation process described below.

The three main radiation sources considered in the design of the cold vacuum drying process are (1) the cask-MCO filled with SNF (the principal source), (2) the process bay local exhaust heating, ventilation, and air conditioning (HVAC) system and process vent system high-efficiency particulate air (HEPA) filters, and (3) the process water conditioning (PWC) system tanks.

A block flow diagram for the CVDF process was developed during the conceptual design phase and revised during the definitive design phase. Throughout the CVDF design process, durations were extracted from the block flow diagrams and dose constraints were established based on the expected dose at each facility location. The current flow diagram, which is documented in the CVDF operations manual (SNF-2356), identifies each step in the cold vacuum drying process and provides an estimate of the time required for each. The time estimates, along with dose rate maps for a typical process bay and the process water tank room, were used as a basis for calculating personnel radiation exposures. The dose rate maps, which are documented in HNF-2850, *Shielding Analysis for the Cold Vacuum Drying Project*, were produced for various elevations in a processing bay and in the process water tank room, from the potential radiation sources within these areas.

By combining information from the block diagram and the contour maps, a model of the CVDF was developed to determine the personnel exposure to radiation that could be expected during operation of the facility. Each activity was evaluated to determine the integrated radiation exposure of the personnel performing the activity. By using this model, the activities that produced the highest personal exposure were identified and investigated for possible exposure reduction as part of the ALARA review process. SNF-4207, *Design Features of the Cold Vacuum Drying Facility to Keep Worker Doses As Low As Reasonably Achievable*, documents the ALARA assessment and includes assumptions and conclusions. Several design changes have resulted from this process.

Specific design features credited with reducing radiation exposure and achieving ALARA program objectives include:

- MCO shipping cask and lid
Shield walls between the bays to reduce exposure from an MCO in an adjacent bay

The tempered water shield ring provides a shield collar over the annular space between the MCO and cask. This eliminated the need to use long-handled tools to perform the operations over the MCO, thereby reducing the durations and total exposure from these tasks.

A local process hood over the manually accessed process ports to capture releases from this potential source and thereby protect operators. Airborne release could occur when MCO connections are made or broken.

A shielded room was provided for collection of the water removed from the MCO and for the removal by the PWC system of the entrained SNF. In addition, major PWC components, such as the ion exchange modules and filter, were designed with integral shielding. These design changes were incorporated to mitigate an increase in the estimated source term in the SNF water that was identified during the design process.

As the previous discussion indicates, the basic principles adhered to in the design of the CVDF were to (1) determine the major contributors to the dose and examine methods for making the process more efficient, (2) provide shielding to reduce the dose, (3) examine the cost-effectiveness of using remote operations, and (4) incorporate confinement features in the design (e.g., special fittings) to confine possible fluid and gas releases. The process was an iterative one that addressed ALARA issues as the design evolved.

Several design measures were incorporated into the CVDF design to reduce the collective dose estimates resulting from maintenance activities. An effort was made to select process equipment with a long service life. In general, components were specified to have a minimum five-year design life. In addition, an effort was made to select equipment with a low frequency of maintenance and calibration. For example, the tempered water and PWC pumps are self-contained pumps with canned rotors and stators that will not need lubrication. During early design activities, one of the major anticipated contributors to maintenance dose was the PWC filter change out. To minimize this contribution, a backwash system was included in the PWC design, which results in an anticipated lifetime of the filter in excess of the facility lifetime.

**B7.5 RADIOLOGICAL PROTECTION TRAINING**

SNF Project requirements and criteria for radiological protection training are described in Section 7.5 of the SNF Project FSAR.
B7 6 RADIATION EXPOSURE CONTROL

A description of SNF Project radiation exposure control measures is provided in Section 7 6 of the SNF Project FSAR. HNF-SD-SNF-DB-003 requires that the Canister Storage Building and CVDF incorporate, as applicable, the following NRC requirements:


- Apply the hourly dose limit of Title 10, Code of Federal Regulations, Part 20, “Standards for Protection Against Radiation,” Section 20 1301, “Dose Limits for Individual Members of the Public” (10 CFR 20 1301), to the design and safety analysis.

- Incorporate control devices for access to high-radiation areas that conform to the requirements of 10 CFR 20, Section 20 1601 “Control of Access to High Radiation Areas”.

For the CVDF, the radiological exposure annual dose criteria of 10 CFR 72 104 have been incorporated into the CVDF design. These criteria apply to design measures to protect any offsite public individual during normal operations and anticipated occurrences. These annual dose equivalent criteria for effluents and direct radiation are 25 mrem (0.25 mSv) to the whole body, 75 mrem (0.75 mSv) to the thyroid, and 25 mrem (0.25 mSv) to any other critical organ. Section 20 4 of HNF-SD-SNF-RD-001, Spent Nuclear Fuel Project Standards/Requirements Identification Document, and Title 40, Code of Federal Regulations, Part 61, “National Emission Standards for Hazardous Pollutants” (40 CFR 61), require that doses received by the maximally exposed individual from emissions of radionuclides shall not exceed 10 mrem/yr. To meet these standards, engineered ventilation systems (as described in Chapter B2 0) have been provided to abate radioactive particulate emissions. Analysis results (DOE/RL-96-110, Radioactive Air Emissions Notice of Construction Cold Vacuum Drying Facility Phase II) indicate that the abated emissions from normal operations will be 0.0042 mrem/yr to the maximally exposed individual. The highest calculated radiation level in the CVDF (HNF-2850) is 100 mrem/yr. Considering the attenuation of this source over the distance to the offsite public individual, the total exposure from direct radiation and anticipated release is considerably less than the 10 CFR 72 104 and 40 CFR 61 criteria. The abnormal events (anticipated occurrences) are identified in Section 3 3 3. That discussion indicates that there are no events that cause a release of radioactive material.

For the CVDF, the criteria for the hourly dose limit to the public, as described in 10 CFR 20 1301, have been incorporated into the design. This hourly dose limit (0.002 rem) is assumed to be from external sources for any unrestricted area during normal operations and anticipated occurrences.
Access to the radiological area of the CVDF (see Figure B7-1) is controlled using one or more of the methods listed and discussed in Section 7.6.2 of the SNF Project FSAR. The potential for the highest dose rates in the CVDF occurs in the process water tank room. To avoid unnecessary and inadvertent doses to personnel, the doors to the process water tank room are provided with locks that can be used to control access. Primarily, access control will be by the use of signs and barricades. Other measures, described in Section 7.6.2 of the SNF Project FSAR, will be used when needed.

At the CVDF, the main sources of shielding for worker protection are the transportation cask, cask lid, the MCO shield plug, and the tempered water surrounding the MCO. See the discussion of MCO and transportation cask shielding in Section 7.6.2 of the SNF Project FSAR.

Permanent shielding is provided by the walls separating the process bays, the process bay support areas, and the administrative area. The PWC ion exchange modules and PWC filter have shielding built into their design. The PWC piping system is designed to allow field installation of lead shielding, if dose monitoring indicates this operational mitigation feature is required. The process water transfer line in the process bays is rinsed after each transfer to minimize retention and buildup of SNF particulate.

Engineering and administrative controls have been included in the CVDF design to minimize airborne radioactivity. The CVDF ventilation system is designed to maintain appropriate building ventilation zone pressure such that air flow is from less to more potentially contaminated areas. In addition, the process bay local exhaust HVAC and process vent system, open-faced process hoods, and process vent connections provide local and system airborne radioactive material control functions. The process bay recirculation HVAC system provides two-stage HEPA filtration of recirculated air within each process bay. This system accomplishes six air changes per hour. The process general supply/exhaust HVAC system is also provided with two-stage HEPA filtration prior to discharge via the CVDF stack. Section B4.4 provides a detailed description of the ventilation system design and confinement philosophy. These systems are used to control any potential airborne material as close to the source as possible and to provide confinement ventilation of the facility. In addition to these installed systems, portable HEPA filter ventilation systems will be used as needed by the radiological job planning of maintenance work.

The ventilation system HEPA filters are accepted by the Washington State Department of Health as the best available radionuclide control technology. Calculations of projected emissions (required by 40 CFR 61) result in the requirement for continuous airborne emission monitoring. The CVDF HVAC airborne emissions monitoring system described in Chapter B2.0, satisfies the 40 CFR 61 monitoring system requirements. This system provides sampling and monitoring functions only and provides no operational control functions.
B7 7 RADIOLOGICAL MONITORING

The radioactive material sampling and monitoring programs conducted internal and external to SNF Project facilities are addressed in Section 7.7 of the SNF Project FSAR. As fuel begins to arrive at the CVDF, a startup surveillance and monitoring program will be implemented to define and characterize exposure rates. The program will include characterization of the MCO receipt, process connections, and drying operations. It will cover grid surveys, exposure rate determinations with initial equipment use, and the change in exposure rates as shielding practices improve.

Several potential areas to be controlled as contaminated areas have been identified at the CVDF. These include small sampling areas associated with the PWC system and small areas on the mezzanine of each process bay where process connections are made.

The CVDF stack emission monitoring instrumentation is described in Chapter B2.0.

B7 8 RADIOLOGICAL PROTECTION INSTRUMENTATION

A summary of the SNF Project plans and procedures governing radiation protection instrumentation is provided in Section 7.8 of the SNF Project FSAR. A description of the types and locations of radiation detection instruments used at the CVDF is presented in Table B7-1 and in Section B2.7.11.

Table B7-1  Selected Survey and Area Monitoring Instruments

<table>
<thead>
<tr>
<th>Quantity</th>
<th>Type/Description</th>
<th>Location</th>
</tr>
</thead>
<tbody>
<tr>
<td>5</td>
<td>Automated personnel monitors (personnel contamination monitors)</td>
<td>1 in each process bay access vestibule</td>
</tr>
<tr>
<td>1</td>
<td>Automated personnel monitor (portal monitor)</td>
<td>1 on the Administrative Building side of door 32 (between the transfer corridor and the Administrative Building)</td>
</tr>
<tr>
<td>10</td>
<td>Area radiation monitors with detectors, logarithmic output, and external dry alarm contacts</td>
<td>2 in each of the four operating bays, 1 in the spare bay, 1 in the process water tank room</td>
</tr>
<tr>
<td>12</td>
<td>Beta continuous air monitors</td>
<td>2 in each of the four operating bays, 1 in the spare bay</td>
</tr>
<tr>
<td>12</td>
<td>Alpha continuous air monitors</td>
<td>1 in the transfer corridor, 1 in the mechanical room, 1 in the process water tank room</td>
</tr>
<tr>
<td>3</td>
<td>Portable neutron monitoring instrument that measures neutron dose from thermal through fast neutrons</td>
<td>Portable</td>
</tr>
</tbody>
</table>
B7 9 RADIOLOGICAL PROTECTION RECORD KEEPING

Radiological protection record-keeping requirements are described in Section 7.9 of the SNF Project FSAR.

B7 10 OCCUPATIONAL RADIATION EXPOSURES

For the CVDF, occupational radiation exposures to the operating staff are estimated based primarily on the manual operations performed in the presence of the radiation field surrounding the cask. The total exposure of CVDF personnel during MCO processing is 62.9 mrem (0.63 mSv) per MCO (including an 11.1 person-mrem [0.11 person-mSv] PWC processing exposure). The ALARA analysis (SNF-4207) assumes 147 MCOs are processed in a year, and therefore, the annual personnel exposure incurred is 9,246 mrem (92.5 mSv). Adding the annual maintenance and calibration dose of 853 mrem (8.53 mSv) and the annual dose of 1,387 mrem (13.87 mSv) from routine surveillance, patrol, housekeeping, and contamination control, gives a total CVDF annual dose of 11,500 mrem (115 mSv). The CVDF staffing plan calls for 25 operators and 15 radiological control technicians to be involved in the accomplishment of these tasks. Thus, the average worker dose per year is 287 mrem (2.87 mSv) total effective dose equivalent, which is well below the 500 mrem (5 mSv) ALARA design goal. In addition, traveling crews associated with MCO transport (15 additional operators) help CVDF personnel in handling the MCO transporter and cask, and there are 35 persons associated with the shared maintenance response crew. When consideration is given to the shared staff, the average operating staff dose is reduced even further. Calculations of the dose rate from neutrons have been performed using conservative methodology and indicate that this dose is negligible.

B7 11 REFERENCES


HNF-3553 REV 0
Annex B — Cold Vacuum Drying Facility


NRC Regulatory Guide 8 8, 1982, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable*, U S Nuclear Regulatory Commission, Washington, D C


SNF-4207, 1999, *Design Features of the Cold Vacuum Drying Facility to Keep Worker Doses As Low As Reasonably Achievable*, Rev 0, Fluor Daniel Hanford, Incorporated, Richland, Washington
Figure B7-1  Radiological Control Boundary
CHAPTER B8 0

HAZARDOUS MATERIAL PROTECTION
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   B8 2 REQUIREMENTS
   B8 3 HAZARDOUS MATERIAL PROTECTION PROGRAM
      AND ORGANIZATION
   B8 4 ALARA POLICY AND PROGRAMS
   B8 5 HAZARDOUS MATERIAL TRAINING
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<table>
<thead>
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<th>Abbreviation</th>
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<tr>
<td>ALARA</td>
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<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
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</table>
B8 0 HAZARDOUS MATERIAL PROTECTION

B8 1 INTRODUCTION

The major provisions of the occupational safety and health program, as the program applies to hazardous material protection for the Spent Nuclear Fuel (SNF) Project are addressed in Chapter 8 0 of the SNF Project Final Safety Analysis Report (FSAR)

B8 2 REQUIREMENTS

The requirements that form the basis for the hazardous material protection program are identified in Section 8 2 of the SNF Project FSAR

B8 3 HAZARDOUS MATERIAL PROTECTION PROGRAM AND ORGANIZATION

The SNF Project has an established, visible, and comprehensive occupational safety and health program. This program is described in Section 8 3 of the SNF Project FSAR

B8 4 ALARA POLICY AND PROGRAMS

While there is no established formal SNF Project as-low-as-reasonably-achievable (ALARA) program for nonradiological hazardous materials, the SNF Project has expanded the classic concept of ALARA (i.e., minimization of radiological exposures) to the application of exposure minimization for hazardous substances and conditions. The SNF Project's policy is described in Section 8 4 of the SNF Project FSAR

Work practices for hazardous material protection and control of chemical exposures, as stated in Section 8 4 of the SNF Project FSAR, will be implemented at the Cold Vacuum Drying Facility (CVDF) using approved SNF Project implementing procedures. The occupational safety and health program will use the additional provisions of Section 8 4 of the SNF Project FSAR. These provisions will also be implemented at the CVDF using approved SNF Project implementing procedures.

Applicable ergonomics considerations, included in DOE Order 5480 10, Contractor Industrial Hygiene Program (Section 9B), under the occupational safety and health program, industrial hygiene subprogram, are included in Chapter B13 0. Ergonomics considerations, along with risk factor control processes and the ergonomics program are described in the SNF Project human engineering program plan (see Section 8 4 of the SNF Project FSAR and SNF-4399)
B8 5 HAZARDOUS MATERIAL TRAINING

Plans and procedures for training SNF Project workers regarding hazardous materials are summarized in Section 8 5 of the SNF Project FSAR

Section 8 5 of the SNF Project FSAR states that SNF Project management provides training professional education, and certification opportunities necessary to support, maintain, and enhance industrial hygiene staff proficiency to meet or exceed U S Department of Energy industrial hygiene training objectives and goals in accordance with HNF-SD-SNF-RD-001, Spent Nuclear Fuel Project Standards/Requirements Identification Document CVDF management is responsible for ensuring that workers assigned to any task involving hazardous materials are trained in the safety and health hazards associated with such hazardous materials Workers will perform only those tasks for which they have received the proper training As the CVDF mission changes, CVDF management will review the training requirements and modify worker training requirements accordingly CVDF management is also responsible for ensuring that retraining is provided within the time allowed by training course requirements

B8 6 HAZARDOUS MATERIAL EXPOSURE CONTROL

Worker safety features at the CVDF are an integral part of facility design and operation The CVDF design encompasses human factors considerations to ensure that operations can be conducted safely SNF Project occupational exposures to hazardous materials and the spread of hazardous material contamination are controlled by a combination of engineered, operational, and administrative controls, and by the use of personal protective clothing and equipment These controls are described in Section 8 6 of the SNF Project FSAR

No significant hazardous materials have been identified at the CVDF as a result of a hazard analysis that was performed and documented in HNF-SD-SNF-HIE-004, Cold Vacuum Drying Facility Hazard Analysis Report, except for the radionuclide content in the multi-canister overpacks (MCOs) Major features of worker protection are presented in Table B3-3 and are categorized by hazard These features are in addition to safety-class or safety-significant features for design basis accidents No safety-significant structures, systems, and components or technical safety requirements have been identified for the CVDF based solely on worker safety considerations

All work activities in the CVDF will receive adequate advance planning so that if potential hazardous materials are identified due to changing conditions in the future, specific precautions will be applied The exposure controls identified in Sections 8 6 1 through 8 6 4 of the SNF Project FSAR will then be implemented at the CVDF using approved SNF Project implementing procedures

Results of detailed CVDF hazard analysis are presented in table form in HNF-SD-SNF-HIE-004 The primary inventory of hazardous material in the CVDF involves the
radionuclide content in the MCOs. Other hazardous material identified by the hazard identification process includes pyrophoric metals and hydrides, oxidizers, potential asphyxiants, hydrogen, diesel fuel, and other flammable or combustible materials. No routine chemical processes will be conducted in the CVDF. Some chemicals, such as those used for equipment decontamination, may be used occasionally. Section 1.2.1 of SNF-2770, *Cold Vacuum Drying Facility Design Basis Accident Analysis Documentation*, states that there are no chemical inventories of safety concern. Table B3-1 identifies hazards by form, type, location, and total quantity.

The hydrogen control strategy includes inerting the MCOs at the CVDF and backfilling them with helium. Technical safety requirements are developed to govern activities associated with potentially high concentrations of hydrogen. CVDF hydrogen management features that provide worker safety are presented in Table B3-3. These features include the heating, ventilation, and air conditioning system, which sweeps hydrogen away from workers and continuous air and oxygen monitors with alarms.

Operating procedures will contain warning or caution statements to alert personnel to operations that could potentially result in oxygen-deficient atmospheres. Personnel performing activities that require the use of inert gas are trained on the appropriate precautions to be taken, including evacuation of the area if the continuous air or oxygen monitor alarm sounds. The monitoring portion of the hydrogen control strategy is described in Section B8.7.

Other hazardous materials at the CVDF (maintenance materials) will be properly inventoried and stored to control hazards inherent to the material, in accordance with SNF Project implementing procedures.

### B8.7 HAZARDOUS MATERIAL MONITORING

Summaries of the hazardous material sampling and monitoring programs that are conducted internally and externally for SNF Project facilities are provided in Section 8.71 of the SNF Project FSAR. The workplace and external monitoring program described in Sections 8.71 and 8.72 of the SNF Project FSAR will be implemented at the CVDF using approved SNF Project implementing procedures. An environmental, radioactivity, and chemical emissions monitoring program, including requirements, is presented in Section 8.72 of the Project FSAR. This program will be implemented for the CVDF in accordance with approved SNF Project implementing procedures.

As discussed in Chapter B3.0, the MCO is prone to hydrogen gas generation from metal oxidation reactions and radiolysis of bound and free water. The hydrogen control strategy includes inerting the MCOs and dilution with inert gas at the CVDF. This inerting and dilution is controlled by the safety-class instrumentation and control system described in Chapter B4.0.
B8 8 HAZARDOUS MATERIAL PROTECTION INSTRUMENTATION

Summaries of plans and procedures governing hazardous protection instrumentation are provided in Section 8 8 and Table 8-2 of the SNF Project FSAR. As stated in Section 8 8 of the SNF Project FSAR, safety and health specialists will determine any need for hazardous protection instrumentation and the number and placement of instruments in the CVDF under normal and emergency conditions in accordance with the requirements stated in Section 8 8 of the SNF Project FSAR. As noted in Section B8 6 and in Table B3-3, oxygen monitors with a local area alarm will be provided as a worker safety feature for several events, including high-pressure helium blowdown, hydrogen explosions, and ion exchange module explosions.

B8 9 HAZARDOUS MATERIAL PROTECTION RECORD KEEPING

The SNF Project has an established document control and records management program. This program is summarized in Section 8 9 of the SNF Project FSAR.

B8 10 HAZARD COMMUNICATION PROGRAM

The hazard communication program applies to the purchase, receipt, transportation use, and storage of hazardous chemicals and products. This program is summarized in Section 8 10 of the SNF Project FSAR. The hazard communication program for the CVDF will be implemented in accordance with the provisions of Sections 8 10 1 through 8 10 6 of the SNF Project FSAR, including hazard posting in work areas, chemical management, chemical labeling, chemical product list, material safety data sheets, information and training. This program will be accomplished in accordance with approved SNF Project implementing procedures.

B8 11 OCCUPATIONAL CHEMICAL EXPOSURES

Predicted annual exposures to workers from hazardous material sources are identified in Section 8 11 of the SNF Project FSAR. Also see Section 8 10 of the SNF Project FSAR for identification of chemical hazard locations, posting, chemical management, and other controls to limit occupational chemical exposure.

B8 12 REFERENCES

DOE Order 5480 10, Contractor Industrial Hygiene Program, U S Department of Energy, Washington, D C


CHAPTER B9 0

RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT
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<th>Description</th>
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B9 0 RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT

B9 1 INTRODUCTION

The essential features of the radioactive and hazardous waste management programs that provide for the safe control, collection, and handling of wastes generated during routine operations at the Spent Nuclear Fuel (SNF) Project facilities are detailed in Chapter 9 of the SNF Project Final Safety Analysis Report (FSAR). This Chapter B9 only applies to waste generated within the Cold Vacuum Drying Facility (CVDF) and those processes and systems designed to deal with that waste. Within the CVDF, bulk water will be drained from the multi-canister overpack (MCO) and processed through ion exchange modules to remove the majority of the radioactivity. There are no chemical inventories of concern for safety analysis considerations.

Potentially radioactive waste from normal operations at the CVDF is generated from:

- Replacement of the process water conditioning (PWC) ion exchange modules
- Replacement of radioactively contaminated high-efficiency particulate air (HEPA) filter elements
- Normal radiation monitoring and decontamination processes
- Waste and debris from failed process equipment
- Contamination-control activities

Small volumes of nonhazardous and potentially hazardous wastes will also be generated (Table B9-1).

B9 2 REQUIREMENTS

The requirements that form the basis for the radioactive and hazardous waste management program are identified in Section 9 of the SNF Project FSAR.

B9 3 RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT PROGRAM AND ORGANIZATION

The facility administrative procedures for solid waste management contain the procedural guidance for the planning, generation, and disposal of generated waste in compliance with applicable requirements. The administrative procedures cover characterization, preplanning, designation, containerization, disposal, and programmatic requirements. A summary of the SNF Project waste management program is provided in Section 9 of the SNF Project FSAR.
Table B9-1 Waste Inventories for the Cold Vacuum Drying Facility

<table>
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<tbody>
<tr>
<td>Solid</td>
<td>PWC IXMs = 61 m³ total for facility operations (radioactive) assumes 7 IXMs utilized</td>
</tr>
<tr>
<td></td>
<td>HEPA filter elements = &lt;2 8 m³/yr assuming 25 units changed every year (potentially radioactive)</td>
</tr>
<tr>
<td></td>
<td>Rags plastic paper wipes solidified decontamination solutions</td>
</tr>
<tr>
<td></td>
<td>anticontamination tapes protective clothing failed equipment vegetation and animal carcasses = &lt;4 m³/yr (140 ft³/yr) (potentially radioactive)</td>
</tr>
<tr>
<td></td>
<td>Fluorescent light bulbs = ~150 bulbs per year (recycled)</td>
</tr>
<tr>
<td></td>
<td>Office trash = &lt;50 m³/yr (nonhazardous nonradioactive)</td>
</tr>
<tr>
<td>Liquid</td>
<td>Waste oils = &lt;4 L (1 gal) total (potentially hazardous and radioactive mixed waste)</td>
</tr>
<tr>
<td></td>
<td>Sewage shipments = &lt;225 000 L/yr (60 000 gal/yr)</td>
</tr>
<tr>
<td></td>
<td>Cooling coil condensate from HVAC = &lt; 150 L/yr (40 gal/yr) (nonhazardous)</td>
</tr>
<tr>
<td>Airborne</td>
<td>There is the potential for airborne emissions to be generated that are radioactive gases and particulates. These emissions have been estimated and the resulting offsite dose is within Title 40 Code of Federal Regulations Part 61 National Emission Standards for Hazardous Air Pollutants (40 CFR 61) standards. The emission control and monitoring systems satisfy state and federal requirements and are permitted</td>
</tr>
<tr>
<td></td>
<td>Helium released to atmosphere = 66 kg</td>
</tr>
</tbody>
</table>

HEPA = high-efficiency particulate air (filter)  
HVAC = heating ventilation and air conditioning  
IXM = ion exchange module  
PWC = process water conditioning

B9.4 RADIOACTIVE AND HAZARDOUS WASTE STREAMS AND SOURCES

The following subsections identify the CVDF waste streams, their sources, and their management.

B9.4.1 Waste Management Process

The goals and policies of the CVDF hazardous and radiological waste management process are described in Section 9.4.1 of the SNF Project FSAR.
B9 4 2 Waste Sources and Characteristics

At the CVDF, both solid and liquid wastes will be generated and gas emissions will be released. The expected generation rates of each CVDF waste stream are listed in Table B9-1. The following subsections describe the sources of the solid and liquid waste summarized in Table B9-1.

B9 4 2 1 Solid Waste Streams and Sources
The major sources of radioactive waste at the CVDF are ion exchange modules that are part of the PWC system. The ion exchange elements remove dissolved radioactive ions and radioactive particles that are suspended in the MCO water. The frequency of changeout of the PWC ion exchange modules is based on ALARA (as low as reasonably achievable) concerns and on maintaining transuranic inventories below the threshold for designation as transuranic waste. In addition, loaded HEPA filters will be generated as a solid waste or potentially radioactive waste. HEPA filter elements will be used for the process water storage tank vents, the process bay local exhaust heating, ventilation, and air conditioning (HVAC) and process vent system, the general supply/exhaust HVAC system, and the process bay recirculation HVAC systems. The filters are not expected to have hazardous materials deposited on them. Solid waste debris is generated during normal operations, equipment maintenance, and any necessary cleanup following abnormal events. Radioactive wastes will be generated from efforts to control contamination and from decontamination necessary for maintenance and repair of contaminated equipment. The debris wastes will consist of contaminated protective clothing, small tools, equipment, vegetation, animal carcasses, and plastic and paper used to control contamination and to decontaminate the shipping casks. At the CVDF, spills of radioactive liquids may result in the use of adsorption and cleanup materials that will be designated as low-level waste. Office trash is also generated on a routine basis but is nonradioactive and nonhazardous.

B9 4 2 2 Liquid Effluents and Sources
The major liquid stream will be the conditioned water that is removed from the MCO and casks during facility operation. Liquid waste oil from maintenance of the vacuum systems is also anticipated on an infrequent basis. At the CVDF, spills of radioactive liquids in the processing bays are contained by the floor sumps, sampled, and disposed of based on the sample results. There will be no sanitary sewer connection to the CVDF. Sewage will be collected in a tank and transported by truck to an offsite sewage treatment facility. A small amount of condensate from the CVDF HVAC cooling coils (less than 40 gal/yr) will be generated during the more humid months.

B9 4 2 3 Gaseous Effluents and Sources
The CVDF is permitted for airborne emissions by DOE/RL-96-110 Radioactive Air Emissions Notice of Construction Cold Vacuum Drying Facility Phase II Section 20 2 2 of HNF-SD-SNF-RD-001 Spent Nuclear Fuel Project Standards/Requirements Identification Document, describes the requirements for clean air permits and their application to the construction, startup, and operation of facilities. See Section 9 4 2 3 of the SNF Project FSAR for general information regarding gaseous effluents and sources. See Chapters B2 0 and B7 0 for information regarding CVDF airborne emission criteria and the
stack monitoring system. Helium will be used in the water removal and drying operations and will be released to the atmosphere after use as part of the air emissions.

**B9 4.3 Waste Handling or Treatment Systems**

The ion exchange modules will be loaded to below transuranic levels so they can be disposed of onsite in the low-level burial ground as low-level category III radioactive waste. Solid radioactive wastes including HVAC HEPA filters (classified as radioactive waste) and other operational wastes (including rags, wipes, plastic, protective clothing, failed equipment, vegetation, and any contaminated animals) will be packaged per the Waste Management Federal Services of Hanford, Incorporated waste acceptance requirements (WHC-EP-0063-5)—described in Section 16 of the SNF Project Standards/Requirements Identification Document (HNF-SD-SNF-RD-001) and in facility waste administrative procedures—and then transported to Waste Management Federal Services of Hanford for disposition. Some of the debris wastes may also be managed under *Comprehensive Environmental Response Compensation and Liability Act (CERCLA)* of 1980 and sent to the Environmental Restoration Disposal Facility for disposal.

Hazardous materials (e.g., fluorescent light bulbs) will be picked up by commercial treatment and disposal facility operators within 90 days of accumulation. Mixed wastes will be transferred to Waste Management Federal Services of Hanford within 90 days of accumulation. Office trash will be picked up and handled as other site nonradioactive and nonhazardous wastes by the site contractor. Office waste (e.g., copier paper) will be recycled to the extent possible.

The PWC water will be taken back to the K Basins in a tanker truck and managed with the other water in the basins. Waste oils are expected to be mixed radioactive and hazardous wastes and will be managed by Waste Management Federal Service of Hanford per approved waste acceptance and management requirements.

Sewage will be sent to an offsite sewage treatment facility. The condensate from the CVDF cooling coils will be collected for sampling and disposal.

Airborne wastes are filtered by HEPA filters and then released per the air permit (DOE/RL-98-30) as noted in Section B9 4.2.3.

Fire suppression water is collected in the effluent drain system sumps in each processing bay and then is automatically directed to the covered retention basin located west of the facility. The contents of this basin are sampled and appropriately disposed of based on the results of the sample.
REFERENCES


Comprehensive Environmental Response Compensation and Liability Act (CERCLA) of 1980
42 U S C 9601, et seq


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INITIAL TESTING, IN-SERVICE SURVEILLANCE,
AND MAINTENANCE
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<td>Cold Vacuum Drying Facility</td>
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<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
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B10 0 INITIAL TESTING, IN-SERVICE SURVEILLANCE, AND MAINTENANCE

B10 1 INTRODUCTION

Essential features of the initial testing program, the operational readiness review, the in-service surveillance program, and the maintenance program implemented at Spent Nuclear Fuel (SNF) Project facilities are described in Chapter 10 0 of the SNF Project Final Safety Analysis Report (FSAR). Cold Vacuum Drying Facility (CVDF)-specific features of these programs are described in this Chapter B10 0.

B10 2 REQUIREMENTS

The requirements that form the basis for the initial testing program, the operational readiness review program, the in-service surveillance program, and the maintenance program are identified in Section 10 2 of the SNF Project FSAR.

B10 3 INITIAL TESTING

The SNF Project initial testing program ensures the operability of equipment and facilities before facility operation. Project details of this program are provided in Section 10 3 of the SNF Project FSAR. Other than a first-article testing program, no special testing requirements prior to startup activities have been identified for CVDF structures, systems, and components.

B10 4 IN-SERVICE SURVEILLANCE PROGRAM

The SNF Project in-service surveillance program is designed to maintain the integrity of facility systems and to ensure that systems perform their function of protecting the health and safety of the public, workers, and facility staff by preventing or mitigating accident consequences. Details of this program are provided in Section 10 4 of the SNF Project FSAR. The technical safety requirements that include the surveillance requirements for the CVDF are described in Chapter B5 0 and in HNF-3673, Cold Vacuum Drying Facility Technical Safety Requirements.

B10 5 MAINTENANCE PROGRAM

The maintenance program for the SNF Project facilities is conducted in accordance with DOE Order 4330 4B, Maintenance Management Program, which provides the general policy and objectives for establishing cost-effective maintenance and repair programs for U S Department of Energy property. The maintenance program will incorporate the results of considerable project...
Annex B — Cold Vacuum Drying Facility

and subsystem vendor interface activities aimed at ensuring an acceptable design and acceptable operating practices relative to reliability, availability, and maintainability.

Policies and procedures are in place to effectively manage SNF Project facility maintenance activities. Section 10.5 of the SNF Project FSAR summarizes the maintenance policies and procedures that are implemented at the CVDF and also describes the process used during the development of the detailed maintenance plan. The structures, systems, and components that are important in the mitigation and prevention of analyzed accidents, and included in the formal CVDF maintenance program, are identified in Chapters B3.0 and B4.0. The design and length of service of the CVDF result in no code requirements for in-service testing.

The basis for operation of the CVDF is to safely vacuum dry K Basins SNF over a period of three years. Operational tasks and drying cycle times have been defined by project design authorities and are discussed in SNF-2356, 

_Spent Nuclear Fuel Project Cold Vacuum Drying Facility Operations Manual_. The planned CVDF drying cycle includes a retest of the main process systems prior to each multi-canister overpack (MCO) drying cycle.

Operational design information has been used as input to a process system design basis capacity study of the SNF Project (HNF-SD-SNF-RPT-011). This study used a summary-level model of major SNF Project facilities to determine the impact of facility interactions on the overall time to complete fuel removal operations. The results indicated that a facility containing four vacuum drying stations satisfies the project goal of completing operations in three years. The study also shows that the fourth drying station will have a low usage factor. An operating efficiency of 70% was used for each drying station, included in the 70% figure was an approximate 20% allowance for equipment breakdown and repairs.

HNF-SD-SNF-DRD-002, _Cold Vacuum Drying Facility Design Requirements_, includes maintainability and availability requirements for the CVDF and specifies a five-year minimum design life for process components and subsystems. Redundancy of major process components is an additional CVDF requirement (HNF-SD-SNF-DRD-002). Operations, startup, and maintenance personnel participated in the design process through performance of design reviews and participation in a failure modes, effects, and criticality analysis, as described in HNF-2576, _Cold Vacuum Drying Facility Phase I FMEA/FMECA Session Report_. Operations also participated in the first-article testing program and provided comments regarding operability and maintainability. Proposed changes resulting from these reviews have been documented, presented to the design organization, and evaluated in subsequent design activities. First-article testing used a full-scale test setup to verify CVDF process design and component performance, to determine drying cycle characteristics including heat-up and cool-down cycles, and to validate the thermal model.

The design organization has included in sub-component purchase requests requirements that vendors provide maintenance recommendations for their components and subsystems based on past performance and failure history. These recommendations are to include spare parts inventories and requirements for special maintenance tools. Since most of the process equipment
design requires commercially available equipment that is provided as skid-mounted packages, vendors have been required to provide maintenance accessibility and testability information Design reviews to date have identified no high-maintenance components First-article testing provided another way to obtain operability, maintainability, and testability input Results from first-article testing have provided numerous design improvements as described in HNF-3342, Hanford Spent Nuclear Fuel Project Cold Vacuum Drying Facility First Article Initial Test Finding Report, and HNF-4057, Cold Vacuum Drying Proof of Performance (First Article Testing) Test Results

In summary, considerable interface has occurred between organizations contributing to the design of the CVDF relative to availability, operability, and maintainability and the results have been considered in the present design This interchange of information will continue as the facility progresses into the startup testing and operational phases

The SNF Project has completed failure mode and effects analyses and fault tree analyses as part of the FSAR hazards assessments, which includes evaluating routine maintenance activities There are no maintenance activities identified other than the specific technical safety requirement surveillance items required to remain in the authorization basis The structures, systems, and components are designed to place the MCO in a safe and stable state without operator action Preventive maintenance and calibrations are utilized to provide continued operability of facility systems and components

The maintenance plan to be described in the CVDF maintenance implementation plan, which will be submitted to the U S Department of Energy, Richland Operations Office for approval, will implement a graded approach to maintenance activities based on sound engineering judgement and knowledge of the facility and will take into consideration the following items

- The design life for the process systems is a minimum of five years, whereas the intended operational life of these systems is three years This short operating life does not warrant an extensive predictive maintenance program but would justify a simple maintenance history program

- In general, most of the process equipment designs require commercially available equipment that is provided as skid-mounted packages that have been tested as part of the first-article test program

- The four-drying-station design allows for considerable downtime for repairs This built-in flexibility and increased availability of equipment for maintenance will allow time for procurement of unanticipated spares and will tend to minimize the size of the required maintenance support
The operational procedures will include a requirement to functionally test the assigned drying station before receiving an MCO at the CVDF. Maintenance problems identified by functional testing will be resolved without interfering with MCO processing.

- Process systems (i.e., process water conditioning, vacuum purge system) have an appropriate level of redundancy that will lead to increased overall availability.

- Utility systems (i.e., instrument air, helium supply, heating, ventilation, and air conditioning) have an appropriate level of redundancy that will lead to increased overall facility availability.

B10 6 REFERENCES


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### LIST OF TERMS

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<td>CVDF</td>
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<td>FSAR</td>
<td>final safety analysis report</td>
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<td>HEPA</td>
<td>high-efficiency particulate air (filter)</td>
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<td>multi-canister overpack</td>
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B11 0 OPERATIONAL SAFETY

B11 1 INTRODUCTION

Features of the Spent Nuclear Fuel (SNF) Project conduct of operations program and fire protection programs are described in the following sections and in Chapter 110 of the SNF Project Final Safety Analysis Report (FSAR).

B11 2 REQUIREMENTS

The requirements that establish the basis for conduct of operations and general aspects of operational safety are identified in Section 112 of the SNF Project FSAR.

B11 3 CONDUCT OF OPERATIONS

"Conduct of operations" is a set of principles that establishes an overall philosophy for achieving excellence in the operation of the SNF Project facilities. SNF Project application of conduct of operations principles is described in Section 113 of the SNF Project FSAR.

B11 4 FIRE PROTECTION

The SNF Project facilities' fundamental fire protection programs are addressed in Section 114 of the SNF Project FSAR. The elements of the fire protection program that are specific to the Cold Vacuum Drying Facility (CVDF) are described in the following subsections. The results of a CVDF fire hazard analysis (FHA), documented in SNF-4268, Fire Hazard Analysis for the Cold Vacuum Drying Facility, are addressed in Section B3 323 and are summarized in the following subsections.

B11 4 1 Fire Hazards

The CVDF FHA (SNF-4268) was developed in accordance with Section 123 of HNF-SD-SNF-RD-001, Spent Nuclear Fuel Project Standards/Requirements Identification Document. The analysis comprehensively assessed the risk from fire at the CVDF to determine that: (1) the potential for occurrence of fire is minimized, (2) a fire would not cause an onsite or offsite release of radiological and other hazardous materials that would threaten the public health and safety or the environment, (3) requirements that will provide an acceptable degree of life safety to SNF Project and contractor workers are in place, (4) property damage from fire and related perils would not exceed an acceptable level, and (5) the safety systems are not damaged by fire. In addition, the FHA evaluates the CVDF design to confirm that DOE Order 5480 7A, Fire
Protection, and DOE Order 6430 1A, General Design Criteria, provide adequate fire protection requirements to achieve nuclear safety equivalency to U S Nuclear Regulatory Commission fire protection requirements.

The FHA (SNF-4268) was prepared to meet the requirements of DOE Order 5480 7A and to evaluate compliance to U S Department of Energy (DOE) fire protection criteria. As required, this analysis addressed the following elements. Proposed changes to the FHA will be screened to the following elements to determine whether safety issues could be created:

- Description of construction
- Protection of essential safety-class equipment
- Fire protection features
- Description of fire hazards
- Life safety considerations
- Critical process equipment
- High value property
- Damage potential: maximum credible fire loss and maximum possible fire loss
- Fire department or brigade response
- Recovery potential
- Potential for toxic, biological, and/or radiological incident due to a fire
- Emergency planning
- Security considerations related to fire protection
- Natural hazards (earthquake, flood, wind) impact on fire safety
- Exposure fire potential, including the potential for fire spread between fire areas

The FHA (SNF-4268) identifies four fire hazards, as follows:

- SNF ignition — The FHA takes credit for the safety-class systems designed to prevent a thermal runaway accident as being effective in the prevention of SNF ignition.

- Hydrogen gas hazards — The FHA concludes that hydrogen gas hazards are adequately controlled by the CVDF design, as analyzed in Chapter B3 4 2 3.

- Fuel oil — SNF-4942, Spent Nuclear Fuel Cold Vacuum Drying Facility Implementation Plan for Fire Hazard Analysis Suggested Actions, takes credit for administrative controls aimed at preventing diesel fuel from igniting the tractor tires during the limited time that the tractor is allowed in the process bay. This approach has been accepted by DOE and the Hanford Fire Department.

- Ordinary combustibles — The FHA takes credit for administrative controls that severely limit the amount and location of transient combustibles in the processing bays and in the auxiliary spaces adjacent to the bays. These controls are described in Section B5 5 7 7. Concentrated fuel loads (e.g., step-off pad combustibles and...
waste) will be isolated with fire-rated enclosures or containers. Limiting combustibles such as those associated with step-off pads, eliminates sources of ignition for the transporter tires, which are the major heat source for most process bay fire scenarios.

The CVDF consists of three adjoining areas: the process bays, the process bay support areas, and the administrative area. The FHA (SNF-4268) divides the facility into two fire areas: the administrative area and a combined process bay and process bay support area. Analysis of the consequences and probabilities of a worst-case fire for the CVDF indicates that the maximum possible fire loss would be realized in a fire in a process bay and process support area. An operations support area fire was also evaluated, but losses from this fire event were determined to be bounded by a process bay and process support area fire because of the equipment replacement costs in the process bay and restart costs. The total cost of facility cleanup and equipment and structure replacement of a process bay and process bay support area fire would be about $77 million.

The FHA (SNF-4268) describes the effects of postulated fires on safety-class equipment. Fires as small as those associated with the combustibles included in a typical step-off pad can affect this equipment, primarily due to temperature affects on the wiring. Fires of concern are located in the process bay fire area because of the location of the safety-class equipment, instrumentation, and the process bays. Temperature effects on this equipment can affect the ability of the equipment to perform its processing functions, but detection of off-normal conditions by the local safety-class instrumentation and control system in each process bay will cause an automatic safety-class instrumentation and control system trip, placing the affected multi-canister overpack (MCO) in a safe, stable condition. Refer to the FHA for additional detail.

Limiting the amount and location of combustibles as specified in the FHA (SNF-4268) and Section B5 5 7 7, prevents the spread of a fire from one CVDF area to another. No release of radionuclides to the environment from the cask and MCO is identified in the FHA as a result of the maximum possible fire loss event due to the ability of the MCO cask to withstand the effects of the fire. No toxicological or biological consequences resulting from fire are anticipated.

B11 4 2 Fire Protection Program and Organization

The fire protection program for SNF Project facilities is structured and implemented in accordance with the operating contractor's safety management policies, philosophies, and criteria described in Section 11 4 2 of the SNF Project FSAR. CVDF-specific aspects of the fire protection program are described in the following paragraphs.

Sprinklers are located throughout the CVDF and are supplied by a connection to the 100 K service water system. The CVDF fire protection system is classified as general service. All areas of the CVDF are automatically monitored for fire and smoke by ionization-type smoke detectors. The fire alarm features include the transmission of signals to the Hanford Fire...
Department and to local building fire alarm annunciators, and shutdown of the appropriate heating, ventilation, and air conditioning units. Life safety criteria are met as required by NFPA 101, *Life Safety Code*. Life safety in the facility is satisfactory. Refer to Section B2.7.2 for additional descriptions of the fire protection system.

DOE Order 6430 1A requires a minimum of two reliable independent sources of water, and RLID 5480 7, *Fire Protection*, requires that two supply points to a looped fire water grid be provided. An exemption and deviation to the second fire water source requirement has been approved by the U.S. Department of Energy, Richland Operations Office (Sellers 1998).

The FHA, Section 18.0, identifies 12 findings (and recommended corrective actions) dealing with the CVDF design, construction, and response to postulated fires. Several of the findings identify areas where the facility does not comply with DOE Order requirements. The CVDF project has responded to these concerns as discussed in SNF-4942. Five of the concerns have been addressed by design changes that implement the applicable requirements. One concern was closed by Letter 99-SFD-167 (Loscoe 1999) and one finding specified that the FHA be updated, as needed, as the SNF Project progresses. The five remaining FHA findings are:

1. A maximum possible fire loss-type fire as noted in the fire model will cause a major fire loss of this facility. The operations could be shut down for up to 18 months as a result of major building and equipment damage.

2. The high-efficiency particulate air (HEPA) filters lack adequate automatic fire protection systems, 2-hour fire-rated enclosures, and heat detection. RLID 5480 7 requires fire protection of the final exhaust and confinement HEPA filters in nuclear facilities as identified in RLID 5480 7, Section 8.2.e, "Filter Plenum Fire Protection Criteria."

3. The fire modeling indicates that the tractor tire fire will cause unacceptable damage levels to the process bays.

4. The processing bays contain safety-class systems. These safety-class systems are not protected from the effects of a fire event as required by DOE Orders 6430 1A and 5480 7A.

5. RLID 5480 7, Paragraph 8.1.e, requires that fire department standpipe connections be provided in areas that have the potential of radioactive contamination. These standpipes need to be Underwriter's Laboratory-listed or Factory Mutual-approved and installed to allow the fire hoses into the confinement structure without blocking open doors.

The SNF Project's response to these remaining concerns is to implement the performance-based program for combustible control described in Section B11.4.3, and to submit and justify equivalencies or exemptions to address the following:
- The RLID 5480 7 requirement for fire protection of the final exhaust and confinement HEPA filters
- The DOE Orders 6430 1A and 5480 7A requirement to protect safety-class systems from the effect of a fire
- The RLID 5480 7 requirement that fire department standpipe connections be provided in areas that have the potential for radioactive contamination

In addition, the process bay support area rooms will be upgraded to a 2-hour equivalent fire rating (Finding 18-1), and the Hanford Fire Marshall has concurred with an exemption to the requirement for standpipes in the process bays (Finding 18-10)

The effects on facility safety of the FHA findings are addressed in Section 5.3 of that document. All of the findings are related to property loss concerns except for FHA Findings 18 2, 18 4, and 18 10. The issues identified in Findings 18 10 and 18 2 are related to reducing the potential for radiological release during fire conditions. Finding 18 4 addresses the issue that redundant safety-class structures, systems, and components (such as the safety-class instrumentation and control system and safety-class helium system), related to a particular CVDF process bay, are located in the same fire area. This equipment must not be subjected to general area temperatures in excess of 461 °C (115 °F) or radiant exposure to fire in a process bay. Table 3 1 1 1 of LCO 3 1 1, “Safety Class Instrumentation and Control Systems,” specifies a high-temperature control at <95 °F where the safety-class instrumentation and control system isolates the MCO (de-energizes eight normally-closed MCO isolation valves) and initiates safety-class helium system purge. These actions ensure that most safety-class equipment will have shifted to a fail-safe mode prior to general area temperatures reaching 461 °C (115 °F). The remainder of the process bay safety-class equipment has been shown to not experience spurious operations below the 461 °C (115 °F) temperature level. Implementation of the performance-based program for combustible control will ensure that the 461 °C (115 °F) temperature limit is not exceeded for the worst-case fire. Two pieces of safety-class equipment (PT-1*08 and PT-1*10) have a temperature limit of 405 °C (105 °F). The SNF Project prevents exceeding these limits by implementing the performance-based program for combustible control plus the installation of a heat shield on the mezzanine between the accepted transient combustible container and the two pressure transmitters.

The SNF Project’s response to Finding 18-2 is to implement the performance-based program for combustible control, which will ensure that the final HEPA filter plenums and associated filters will not plug, burn, or blow out when subjected to the worst-case fire.

B11 4.3 Combustible Loading Control

Control of combustibles in the CVDF is critical to the safety of the facility, as described in the FHA (SNF-4268) and in Section B3 3 2 3 Section 11 4 3 of the SNF Project FSAR.
summarizes the SNF Project program used to prevent unnecessary combustible loadings in the project facilities. For the CVDF, the quantity of combustibles anticipated to be present in the process bays and their support area is very limited and will include cable, piping, and duct insulation, transient combustibles such as contamination control supplies and waste, and anti-contamination clothing. When an MCO is being processed in the bay, trailer tires will also be present.

Per the FHA (SNF-4268) and Section B5 5 7 7, a performance-based program for combustible control will be established to strictly control, within the facility, the quantity and location of transient combustibles. The FHA (Table 2 as modified by SNF-4942) identifies the maximum allowable quantities of transient combustibles and the allowable locations for the combustibles. Allowable containers for transient combustibles are described in Appendix A of SNF-4942. The transient combustibles controls program is required by technical safety requirement Administrative Control 5 13 and will be implemented through operating procedures, as prescribed in Section 12 1 2 of the SNF Project Standards/Requirements Identification Document (HNF-SD-SNF-RD-001).

**B11 4 4 Fire Fighting Capabilities**

The Hanford Fire Department maintains a training program for fire fighting, fire system testing and maintenance, and building inspections. Fire fighting capabilities that apply to all SNF Project facilities are addressed in Section 11 4 4 of the SNF Project FSAR. CVDF-specific fire response procedures are addressed in the following paragraphs.

All areas of the CVDF are automatically monitored for fire and smoke by ionization-type smoke detectors. Manual pull stations are provided at each facility exit. The fire alarm features include transmission of signals to the Hanford Fire Department via a radio fire alarm reporter box. Fire alarm annunciating devices are provided for occupant notification.

A fire brigade is not required at the CVDF due to the nearby presence of the Hanford Fire Department. The standard response to an alarm condition in the 100 K Area is by the Hanford Fire Department from the 100 Area Fire Station. The 100 Area Fire Station may be relocated—including personnel, trucks, and equipment—to some undetermined location in the 100 K or 100 N Areas. Hanford Fire Department response time from the 100 Area Fire Station to the CVDF is approximately five minutes. A crew from the 200 Area Fire Station will be dispatched simultaneously with an estimated response time of 15 to 20 minutes. Because the 100 Area Fire Station is not manned full-time, the 200 Area response team will be the primary responders at times when the 100 Area Station is unmanned. These are the response times and the responder locations assumed in the FHA (SNF-4268). Refer to WHC-SP-1180, *Hanford Site Emergency Response Needs*, and HNF-SP-1180, *Hanford Site Emergency Response Needs Implementation Plan*, for additional description of the Hanford Fire Department response capabilities. Vehicle access to the CVDF is provided by a paved access road. The Hanford Fire Department is fully staffed, trained, and equipped for emergency response.
Facility personnel are trained on the expected actions to be taken in case of a fire. Personnel are expected to notify the Hanford Fire Department, evacuate the facility, and follow approved fire response plans specific to the facility.

**B11 4.5 Fire Fighting Readiness Assurance**

A prefire plan for the CVDF will be prepared by the Hanford Fire Department prior to facility operations. The FHA (SNF-4268) provides a description of fire-fighting methods that will be employed in the plan.

A summary of SNF Project fire prevention inspections, fire safety drills and exercises and program record-keeping requirements is provided in Section 11 4.5 of the SNF Project FSAR.

**B11 5 REFERENCES**


DOE Order 6430 1A, *General Design Criteria*, U S Department of Energy, Washington, D C


CHAPTER B12 0

PROCEDURES AND TRAINING
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<th>Description</th>
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<tbody>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
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B12 0 PROCEDURES AND TRAINING

B12 1 INTRODUCTION

A description of the essential features of Spent Nuclear Fuel (SNF) Project procedures and training is provided in Chapter 12.0 of the SNF Project Final Safety Analysis Report (FSAR).

B12 2 REQUIREMENTS

The requirements that form the basis for the SNF Project training and procedures programs are identified in Section 12.2 of the SNF Project FSAR.

B12 3 PROCEDURE PROGRAM

SNF Project activities are conducted in accordance with written procedures. A summary of the facility procedures program, including development and maintenance of procedures, is provided in Section 12.3 and its subsections in the SNF Project FSAR.

B12 4 TRAINING PROGRAM

The objective of the SNF Project personnel training program is to provide and maintain a qualified work force for safe and efficient facility operations. A summary of the SNF Project personnel training program—including training development, maintenance of training, and modification of training materials—is provided in Section 12.4 and its subsections in the SNF Project FSAR. Qualification of the Cold Vacuum Drying Facility initial crews will use knowledge-based training presented in the classroom, and skills-based tasks primarily addressed by on-the-job training activities. On-the-job activities for the initial crews will include involvement in dry-run demonstrations and special training evolutions.
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CHAPTER B13 0

HUMAN FACTORS
HNF-3553 REV 0
Annex B — Cold Vacuum Drying Facility

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<tbody>
<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
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<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
</tr>
<tr>
<td>DOE</td>
<td>U S Department of Energy</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
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<tr>
<td>HCl</td>
<td>human–computer interface</td>
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<tr>
<td>HFE</td>
<td>human factors engineering</td>
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<td>HMI</td>
<td>human–machine interface</td>
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<tr>
<td>HSI</td>
<td>human–system interface</td>
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<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
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<tr>
<td>MCS</td>
<td>monitoring and control system</td>
</tr>
<tr>
<td>NRC</td>
<td>U S Nuclear Regulatory Commission</td>
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<tr>
<td>RFP</td>
<td>request for proposal</td>
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<tr>
<td>SCIC</td>
<td>safety-class instrumentation and control</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
</tr>
<tr>
<td>SSC</td>
<td>structure, system, and component</td>
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B13 0 HUMAN FACTORS

B13 1 INTRODUCTION

General features of the Spent Nuclear Fuel (SNF) Project human factors engineering (HFE) process are described in Chapter 13 0 of the SNF Project Final Safety Analysis Report (FSAR) Specific application of the human factors process to the Cold Vacuum Drying Facility (CVDF) is described in this Chapter B13 0

This chapter will show that human factors were considered by the design team throughout the design process for the CVDF Almost all human factor issues have been identified and have either been resolved or are being resolved as design change information becomes available An example of an issue that has not been fully defined is the process for manually moving the mezzanine bridge to the cask multi-canister overpack (MCO) work platform This issue is being followed as final design decisions are made

An assessment team conducted a hazard analysis to identify potential hazards to the CVDF operator, co-located personnel, the public, and the environment (HNF-SD-SNF-HIE-004) Hazards were analyzed based on energy source and material, and established “defense-in-depth or worker safety features” were presented In addition, the final HFE analysis (SNF-4213) outlined possible areas of human error during a task analysis and recommended ways to mitigate the potential human error, however, this HFE analysis was not a human reliability study Chapter 3 0 of the SNF Project FSAR describes the hazard analysis methodology, and results for the CVDF are summarized in Chapter B3 0

In addition, a review of Hazards Analysis of SNF CVD System - Off-Normal Assessment (Ares 1998) was completed The team performing the off-normal assessment identified and assessed operational upsets (e g , valve failures, control upsets, mechanical equipment failures) that potentially could occur during normal operating conditions The team is finishing work on specific pre-approved operational responses that will be needed following an off-normal event to ensure that the facility safety envelope and boundaries are not challenged

The design team developed and revised SNF-2356 Spent Nuclear Fuel Project Cold Vacuum Drying Facility Operations Manual, which contributed to understanding the role of humans in the operation of the CVDF The formality and the extent of the systematic inquiry into the human factors associated with the CVDF were determined based on the extent of human interaction, the system design effort, and the risk associated with human performance failures

The CVDF design is ready to begin the validation and testing phase This phase will provide an opportunity to observe humans in the performance of allocated tasks within the context of full system simulation or operation
B13 1 1 Brief Description of Facility Purpose

The CVDF classification is a hazard category 2 nonreactor nuclear facility. The CVDF purpose essentially is outlined in three parts: (1) to accept casks with inserted MCOs containing SNF, (2) to remove free water from the SNF using a technical process, and (3) to prepare the cask-MCOs for transport to the Canister Storage Building (CSB). The CVDF provides the required technical process systems, supporting equipment, and facilities needed to support this purpose. The cold vacuum drying process involves draining bulk water from the MCOs and subsequent vacuum drying. After vacuum drying, the MCOs are backfilled with helium. Removal of free water from the MCOs is necessary to reduce the potential for fuel-water corrosion reactions that could lead to MCO overpressurization at the CSB. Figure B13-1 shows an overview of the CVDF process.

The CVDF includes four process bays that can each accommodate an MCO on a specially designed trailer that has been shipped from the K Basins. The CVDF also conditions contaminated process water from the MCO for transport back to the K Basin West facility.

Human system interaction within the CVDF mainly focuses on two important areas: the process bay(s) and the control room. The process bay provides direct human access to the cask-MCO and exposes the human operator to some risk. The control room provides the human operator a means for management and control over the cask-MCO processing operation(s). For purposes of the human factors safety basis, the control room is a nonreactor control room. Chapter B2 0 presents a complete facility description.

B13 1 2 Scope

This chapter presents a systematic inquiry of CVDF human factors and defines the CVDF human factors safety basis, as required by DOE Order 5480 23, Nuclear Safety Analysis Reports. Human factors is one of three aspects of facility safety assurance. The other two aspects are design and management. Therefore, the systems approach to design is applied whereby systems, structures, and equipment are built in accordance with the best available design practices. The essential features of the human system interaction relating to facility safety are the primary scope of this chapter, however, human factors are not limited to only safety-class or safety-significant applications but are intended to cover the gamut of CVDF operations. This chapter presents a discussion of the SNF Project incorporation of human factors and HFE into the design and operation of the CVDF. This annex provides additional plans for continuing to incorporate HFE as the design enters the final test and validation phase. This annex uses DOE-STD-3009-94, Preparation Guide for U S Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports, for additional guidance and format description.
B13 1 3 Objectives

The CVDF is in the final design phase with all significant human–machine interfaces (HMIs) defined, requirements produced, and most of the hardware and software procured mocked-up, or built. The objectives of this chapter are to:

- Review, incorporate, and complete the HFE efforts accomplished earlier in the design stage during development of the preliminary human factors analysis, as documented in HNF-SD-SNF-SAR-002, Safety Analysis Report for the Cold Vacuum Drying Facility, Phase 2, Supporting Installation of Processing Systems.

- Review and incorporate applicable customer (U.S. Department of Energy [DOE]) comments made during review of the preliminary human factors analysis documented in HNF-SD-SNF-SAR-002.

- Demonstrate a systematic inquiry into the importance of safety pertaining to reliable, correct, and effective HMI, including an HFE review of the safety-related structures and components that include HMI.

- Describe applicable human interaction in areas of surveillance, maintenance, normal, off-normal, and emergency operations, and summarize the results of the current facilities hazard analysis regarding human operations.

- Ensure appropriate optimization of the design to enhance reliable human performance, while reducing the risk of human error.

- Review the findings from SNF-2825, Spent Nuclear Fuel Project Cold Vacuum Drying Facility Human Factors Engineering Analysis Results and Findings.

- Include the results from SNF-4213, Final Human Factors Engineering Report for the Cold Vacuum Drying Facility Analysis, Results and Findings (which addresses the issues identified in SNF-2825).

- Make recommendations for subsequent human factors support, as appropriate.

B13 2 REQUIREMENTS

This section details the design codes, standards, regulations, and DOE orders applicable for establishing the safety basis of the SNF Project facilities. The intent is to provide only the requirements that are specific for HFE and pertinent to the HFE safety analysis. This section does not provide a comprehensive listing of all the industrial standards, codes, or criteria.
B13 2 1 U S Department of Energy Orders and Standards

The following DOE orders and standards are applicable to the human factors safety analysis:

- **DOE Order 4330 4B, Maintenance Management Program**, provides consideration of written procedures to guide applicable operations and maintenance.

- **DOE Order 5480 19, Conduct of Operations Requirements for DOE Facilities**, considers the allocation of control functions for operations and maintenance to humans and machines. This order also considers staffing and the qualifications of personnel for operations and maintenance as well as written procedures to guide operations and maintenance.

- **DOE Order 5480 20A, Personnel Selection Qualification and Training Requirements for DOE Nuclear Facilities**, provides consideration of staffing and qualification and training of personnel for operations and maintenance.

- **DOE Order 5480 23, Nuclear Safety Analysis Reports, Attachment 1, Section 3 c**, provides the rationale for upgrading the safety analysis requirements to include HFE. Topic 14 of this order details provisions affecting the systematic inquiry of the importance to safety of reliable, correct, and effective HMI. Topics 12, 13, and 15 of this order provide consideration of staffing and qualification and training of personnel for operations and maintenance.

- **DOE Order 6430 1A, General Design Criteria, Section 1300-12**, provides general criteria for incorporating HFE considerations into the system design process. It also provides HFE considerations for system and component displays, controls, alarms, labeling, and communications that are generally applicable to a wide range of human-machine systems. This order also provides considerations for the work environment for personnel, including such matters as ventilation, lighting, noise control, workspace layout, and equipment design and layout.


- **DOE-STD-3009-94, Preparation Guide for U S Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports, Chapter 13**, provides guidance in providing information that will satisfy the applicable HFE requirements of DOE Order 5480 23.
B13 2 2 U S Nuclear Regulatory Commission Requirements

- NUREG-0700, *Human System Interface Design Review Guideline*, provides HFE guidelines for use in reviewing a specific human—system interface (HSI) design implementation. This ensures that the HSI supports safe, efficient, and reliable personnel task performance. Aspects of the HSI design review process that are important to the identification and resolution of human engineering discrepancies are also described.

- NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Plants*, Section 18, "Human Factors Engineering, Control Room" provides guidance to staff reviewers in the Office of Nuclear Reactor Regulation who perform safety reviews of applications to construct or operate nuclear power plants. This document is included in the requirements list to provide the SNF Project with a review of the U S Nuclear Regulatory Commission (NRC) guidance on DOE Order 6430 1A, Section 1300-12 4. NUREG-0800 assists in the identification of appropriate additional NRC guidance for the SNF Project facilities. HFE design support. More specifically, NUREG-0800 considers the complexity differences between a nuclear reactor control room and the control room being provided by the SNF Project.

B13 3 HUMAN FACTORS PROCESS

HFE is important in ensuring the safe operation of the CVDF. This section summarizes the human factors process for systematically evaluating the importance of human factors in CVDF safety. The overall purpose of this process is to ensure that the HSI with the CVDF supports safe, efficient, and reliable personnel task performance. The objective is to identify and resolve human engineering discrepancies that could adversely affect CVDF safety. This section also presents a discussion of the means by which the SNF Project has incorporated HFE into the current CVDF design and operation.

The incorporation of HFE considerations into the CVDF facility design and operational processes ensures enhanced human contributions to operational success while reducing abnormal situations resulting from human error. NUREG-0700 provides insight into processes for systematically evaluating the insertion of the human into the CVDF system. Relevant HSIs include all alarms, displays, controls, labeling, job performance aids, workstation and workplace layouts, and physical accessibility as well as environmental conditions such as lighting, noise, temperature, humidity, and protective clothing.

For the CVDF, the design engineering goal is to reduce HSI to a very low level. This decision reduces the chances for human error, increases the efficiency of the system, and reduces the personal risk of injury and illness to the operators. Allocating most of the process functions to automation and software technology, the principal design purpose for humans in this system is preparing the cask—MCO for processing (bay activity) and monitoring the process through
human–computer interfaces (HCI), with occasional permission commands that tie the human into the normal operating process. To achieve this goal, the human factors process involves the following strategies:

- Conceptualizing the design and initially allocating broad-based functions to either the human or machine
- Identifying and creating design requirements documents
- Developing the system safety-class requests for proposals (RFPs) that include HFE criteria
- Developing system design descriptions that include operator and maintenance personnel actions and HFE considerations
- Examining both system responses and operator actions during off-normal and emergency conditions
- Developing maintenance, training, and staffing based on system design descriptions and allocations of human functions
- Developing full-scale mock-ups of proposed designs including the CVDF process skid, cask–MCO, process connectors, and human-computer software
- Providing CVDF cognizant engineer formal design reviews at design stages (e.g., preliminary [30% complete] and final [90% complete]) to ensure incorporation of HFE criteria, as applicable, completion of HFE comprehensive review checklists to compare the design structures, systems, and components (SSCs) against standard human error design criteria, principles, and practices
- Resolving human engineering discrepancies as revealed during the human error comprehensive checklist review, using additional resources involving actual CVDF building and vendor-submitted components and tools to further evaluate human interfaces
- Providing for continuing human error review of design changes as changes occur and incorporation of applicable changes
- Reviewing programmatic concerns, if any
- Verifying and validating the entire HFE effort using step-by-step procedures just before operational startup of the entire CVDF system
- Reporting final results of the human factors evaluation process
Figure B13-2 shows the human factors process integrated with the total design effort

B13.3.1 Process Background

HFE is the application of knowledge about human performance capabilities and behavioral principles to the design, operation, and maintenance of human-machine systems. Application of this knowledge ensures that personnel can function at their optimum level of performance with a system optimally designed for human use. Though optimum design is the goal, humans do have the ability to adapt to a less-than-optimum design, however, exceeding human adaptability limitations will cause the system to become inefficient and potentially unsafe.

CVDF HFE considerations are commensurate with the following:

- Planned SNF Project mission
- Hazard category 2 classification
- Complexity of the CVDF
- Level and type of human interfacing with the SSCs and the SNF Project processes

The extent of the human interaction, the overall level of the current design effort, and the risk associated with human performance failures (human error) guide the formality and the extent of the systematic inquiry into the human factors associated with the CVDF. Emphasis is on the HMI required for ensuring the safety function of SSCs that are important to safety. Throughout the design effort, the evidence demonstrates that human factors are considered in facility operations where humans are relied on for preventive actions (e.g., surveillance and maintenance activities during normal operation) and for operator mitigative actions during off-normal and emergency operations.

Chapter B10.0 and Chapters 12.0 and 17.0 of the SNF Project FSAR discuss the staffing and qualifications of personnel, the required operator and maintenance personnel training, and the written procedures to guide operations and maintenance. The human factors process also reviewed this material for HFE adequacy. Equipment design effects on maintenance efficiency are consistently shown to be greater than such personnel variables as behavior or training. A description of HFE impacts during design for maintainability efforts are described in SNF-4399, Spent Nuclear Fuel Project Human Engineering Program Plan (HEPP).

Per SNF-4399, engineering of the HMI aspects of design should reflect the following HFE/Erg considerations, annotated below, as applicable. In addition, these features are captured in the HFE/Erg design review through use of the INEL-95/0117, Human Factors Engineering Checklists for Application in the SAR Process, located in the SNF Project Office. INEL-95/0117 also captures elements of UCRL-15673, Human Factors Design Guidelines for Maintainability of Department of Energy Nuclear Facilities, for example, accessibility requirements. Examples of the SNF-4399 HFE/Erg features considered for design are...
Compatibility of the design, location, and layout of controls, displays, work spaces, maintenance accesses, stowage provisions, etc., with the clothing and personal equipment to be worn by personnel operating, riding in, or maintaining DOE systems or equipment

Design features to assure rapidity, safety, ease and economy of operations and maintenance in normal, abnormal, and emergency maintenance environments

Adequate space for personnel, their equipment, and free volume for the movements and activities required during operations and maintenance tasks under normal, abnormal and emergency conditions

Efficient arrangement of operation and maintenance workplaces, equipment, controls, and displays

Adequate natural or artificial illumination for the performance of operation, control, training, and maintenance

Satisfactory working conditions including pressure, temperature, humidity, vibration, and acoustic considerations

Satisfactory remote handling provisions and tools

The human-machine interface shall represent the simplest design consistent with functional requirements and expected service conditions. Personnel shall be capable of operating, maintaining, and repairing equipment in their operational environment with a minimum of training

Design will include minimization of potential human error in the operation and maintenance of the system, particularly under emergency or non-routine conditions

Equipment and systems shall be designed so that they can be easily maintained. Considerations for this should include time, cost, safety, reliability, minimum expenditure of support resources, and radiological exposure. To ensure that costly maintenance or redesign are avoided, ease of maintenance shall be designed into facilities, systems and equipment

The essential HSI analyses are complete with the recommended human error design modifications provided to the design authorities through methods such as HFE analysis reports, tracking of human engineering discrepancies, and direct contact between human factors personnel and the design authorities. Any significant changes affecting the operators will be evaluated for human factors effects and reported in subsequent updates to this annex. The HMI detail review is provided in Section B13.4 and in SNF-4213.
As described in SNF-4213, the CVDF is well designed from the HFE perspective. The CVDF process and equipment designers have given much consideration to HFE issues brought to the attention of the designers were handled in a professional manner and, after design trade-off evaluation and cost-benefit analysis, were incorporated as deemed appropriate.

Applicable HFE requirements and criteria found in DOE orders and standards have been applied to the CVDF design. Four outstanding concerns were identified and investigated for possible improvement:

- Manual cask-multi-canister overpack (MCO) bolt removal and replacement
- Manual water hookup at the bottom of the cask and associated newly designed tools
- The bridge structure to cross from the mezzanine to the cask-MCO work platform
- The MCO process connectors connection to MCOs

The SNF Project is addressing these concerns. For example, the manual water hookup at the bottom of the cask is being design with a lighter, shorter extension and with an improved connector that provides a better leakproof connection and improves the ease of inserting and hooking up the hose.

Two focus areas identified in the Preliminary HFE Analysis report (SNF-2825) were the monitoring and control system (MCS) and safety-class instrumentation and control (SCIC). These areas were studied in detail and found to meet or exceed HFE checklist criteria, as reported in SNF-4213. The MCS comprises a total of three control room computers, two operator workstations, and one engineering workstation. A fourth computer station is provided as a supervisor station in the manager’s office. The main programmable logic controller control panel is also located in the control room and includes a local alarm and silence button. Four remotely controlled input/output modules with termination panels are located in the bays—one is in the mechanical room, and one is in the PWC tank room. Human interaction uses a keyboard, mouse, and monitor. All CVDF processing systems information is presented to the operator at the MCS. The MCS in association with the SCIC provides the operator control over the processing system, throughout various modes of operation.

The MCS HCI is heavily dependent on object-oriented color graphic displays on computer screens. All central processing units are connected on a local area network, which allows simultaneous access to the control system from multiple graphic displays. The system design description for the MCS is contained in SNF-3090, Monitoring and Control System. SNF-2408, System Design Description for the Cold Vacuum Drying Facility Monitoring and Control System, provides for the CVDF MCS software engineering. Also, HNF-2058, Specification for Cold Vacuum Drying (CVD) Project Monitoring and Control System Computer Software Requirements Specification, supports the software design description (SNF-2408) by providing specific requirements in the software engineering (e.g., the exact meaning of colors used in the...
graphical displays) "IEEE Standards Collection: Software Engineering" (IEEE 1997) is also consulted by the software engineer(s) to provide additional guidance, as applicable. These references include elements of HFE that are applied across industrial computer-based systems.

During operations, the human operator interfaces with the SCIC system through the mode switch panel in the control room and the safety-class annunciator. Numbers as well as text describing the mode position were added to the mode switches to provide two indications of the required switch position(s). For example, the procedure will state to select "Dryng Mode for Bay 2, HS-6000." The annunciator provides standard industrial format for alarm receipt, acknowledgment, and reset.

Overall, the design of the CVDF will meet HFE requirements. No significant changes are anticipated during the system test, verification, and validation phase. Human factors emphasis on the four items listed above, must continue as the design evolves into the system validation phase. In reference to the Preliminary HFE Analysis report (SNF-2825), the CVDF was found to be in compliance with 99% of the HFE criteria that it measured.

The following information is also provided.

**Crane** The overhead crane meets all applicable checklist items. Crane design and human interface (control) are in accordance with standard industry specifications, and control design is typical of that used elsewhere on the Hanford Site for similar operations. No human-interface concerns with the CVDF crane were identified.

**SCIC** The HMI meets all applicable checklist items. The SCIC panel design was studied for HFE inputs and found to be acceptable. The SCIC alarm system is differentiated from the MCS alarm to indicate to the operator which alarm system is activated. The SCIC is virtually automatic in operation. Main human interfacing is testing the system and operationally placing the mode switch in the proper position during the phases of the process cycles.

**Process Skid** Evaluation of the mock-up indicated some minor concerns with the gauges and with accessibility. The procurement specification addresses these concerns and indicates that HFE must be considered (e.g., all gauges must be designed to be read by a person standing on the floor). It is possible for maintenance personnel to access the necessary functions for routine maintenance. Labeling on the skid is called out in detail in the design/procurement specification and meets or exceeds HFE standards.

**MCO and Equipment** The process connectors "T-handle" mechanisms were evaluated by the operators at the mock-up facility. It was decided that the T-handle mechanisms were superfluous and a hindrance. The operator simply maneuvers the process connectors while holding onto the connector piping itself. The piping serves as a "handle," and there appears to be no engineering problem with using the piping as a "handle." The process hood is maneuvered into place using the overhead crane, the operator manually guides the hood into proper position. Other concerns with the cask are as follows.
• The 12 bolts must be manually un-torqued and re-torqued with 300 ft-lb (±10%) of force. The plan is to use a torque multiplier to assist in this task. The torque multiplier will use a ratio of 6:1 to bring the physical human effort down to 50 ft-lb. The torquing sequence for the bolts must be followed exactly according to engineering requirements. Thus, this job will require two persons working with six bolts per person per cask. Weighing considerations such as cost-benefit and duty time with this task, it was determined that the task can be successfully performed as planned. Consulting MIL-STD-1472E (Department of Defense Design Criteria Standard, "Human Engineering") and using the high friction case (coefficient of >0.9), it is possible that a manual push/pull force of 70 lb may be applied with both hands.

• The other concern involves the method of removing and installing both the lower cask access plate and the water hookups, using two tools. The tools to do these tasks are being redesigned to shorten their lengths. The redesign strikes a balance between the ALARA requirements and the ease of use of the tools.

MCS The MCS is located in the Central Control Room. The design of screen presentations and alarms meets applicable HFE criteria presented on the checklist review. The HCI appears well designed using the computer software requirements specification (HNF-2058). NUREG-0700 design criteria have been incorporated as applicable. The design provides for two operators in the Central Control Room, each monitoring two bays using video display terminals (VDTs). All systems can be monitored through the computer systems in the control room. The operators will be able to interact with the systems via the VDTs (mouse and separate keyboard based). Operators are alerted to off-normal conditions and take the appropriate actions. The Central Control Room work environment is planned for installation of fully adjustable ergonomically designed seating and fully adjustable computer workstations, both for the keyboard and for the monitor platforms. Operators will be able to adjust the VDT workstations to accommodate their particular needs. Operators using the mock-up facility have interacted well with the design engineers to make changes in the system design, as appropriate, and to allow increased operator efficiency in using the system. Information overload is being considered and reduced, as appropriate.

One of the primary human concerns with the MCS is the efficiency of the MCS. The system will run virtually automatically in the normal mode of operation. The human provides the permission signal for the system to move from one sequence to another. This is a relatively inactive mode of operation for the human, so other administrative considerations may need to be included (to introduce variability in operator tasking). Also, shift scheduling will be important to manage.

The work environments are well designed regarding ventilation, lighting, noise, and temperature. The facility design criteria meet applicable codes and call out specific temperature and humidity criteria with seasonal adjustments. These are well within the comfort zone for the personnel working within the facility. Lighting calculations and examination of the Central...
Control Room data were completed. These calculations appear to meet illuminance requirements for the VDT work surfaces. Since the lighting calculations were completed, a third ceiling fixture is being added to the Central Control Room. New calculations may need to be completed. One possible concern is the presence of glare on monitor screens. This may be examined during the test and validation phase, and glare filters may be used on the computer monitor screens. Lighting control (brightness) should be an operator function.

**B13 3.2 Human Factors Engineering Checklist Procedure**

This chapter describes the process of applying the preferred HFE checklist (INEL-95/0117) during the analysis. The checklist process methodology supplements the overall human factors process (described above) during evaluation of the CVDF SSCs for HFE considerations. The checklist process analysis follows the following logical course:

- Becoming familiar with CVDF SSCs
- Reviewing the facility hazards evaluation (HNF-SD-SNF-HIE-004)
- Selecting the HMI to be evaluated based on the systems that fall into one of two categories (1) safety class SSCs, or (2) pertinent to the effective operation of the CVDF
- Conducting a tabletop task analysis to determine the task performance, the sequence of operations, and the maintenance activities, reevaluation with the mock-up and identified tools
- Interviewing design authorities and other cognizant project personnel to determine the current level of design and anticipated design efforts
- Reviewing the design requirements process, pertinent subcontract RFPs, and overall facility design process for inclusion of human factors requirements, particularly design requirements called out in DOE Order 6430 1A, Section 1300-12, HFE
- Applying the appropriate HFE checklists to the existing and final SSCs and design requirements documents
- Sorting checklist evaluation results into one of four categories (1) complied, (2) not complied, (3) not applicable, and (4) to-be-determined (usually the case if design is not complete or pending change)
- Resolving noncompliance issues by reapplication of the HFE checklist
Charging the design authorities with commitments to follow up with the "to-be-determined" cases to ensure their inclusion in further design efforts.

- Reviewing programmatic issues (staffing, maintenance, procedures and training) and resolving any programmatic concerns.

- Reporting the results of the HFE evaluation process both in this chapter and in the detailed HFE report.

HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward Additional NRC Requirements, Item 9, calls out a review of the NRC guidance in NUREG-0700 and NUREG-0800 against DOE Order 6430 1A, Section 1300-12 4, and draft DOE-STD-1062-94 Ergonomics and Human Factors Engineering Design Criteria Volume 1, to identify appropriate additional NRC guidance for design of the CVDF. The goal of NUREG-0800 is to ensure that the design of a nuclear power plant control room complies with accepted HFE principles, focusing on the following areas:

A. Control room work space
B. Workspace environment
C. Annunciator warning systems
D. Controls
E. Visual displays
F. Auditory signal systems
G. Labels and location aids
H. Process computers
I. Panel layout
J. Control-display integration

In reviewing NUREG-0800 against DOE Order 6430 1A, it is apparent that the applicable items of interest (A through J) above, are also represented in the DOE Order 6430 1A requirements. The DOE requirements are then captured in INEL-95/0117 Human Factors Engineering Checklists for Application in the SAR Process, which uses the design criteria from various sources, including NUREG-0700 and DOE-STD-1062-94, as called out in HNF-SD-SNF-DB-003, Item 9. Using a graded approach and focusing on significant human-machine interfacing, the systems listed in the SNF Project FSAR and this facility FSAR Annex were deemed appropriate for HFE checklist evaluation using applicable INEL-95/0117 checklists (which cover items A through J above in great detail) for each system.

Figure B13-3 depicts the human factors checklist procedure.
B13 3 3 Engineering Design and Analysis

Engineering design encompasses the design process and analysis in support of both original design and design change activities. Design preparation is in accordance with criteria established in DOE Order 6430.1A, Section 1300-12, where human factors are considered through the system development process (planning, requirements analysis, system design, and system test and evaluation). Pursuant to DOE Order 6430.1A, Section 12.4, the design, operation, and maintenance of the CVDF consider appropriate human factors technology to improve human performance through enhancements in the work environment and HMI. HFE applies at a level commensurate with the safety importance, complexity, and degree of HMI of the SSCs.

In completing the final design for the CVDF (HNF-SD-SNF-DRD-002), human factors design analyses encompassed design criteria such as operational aids, the control and display relationships, visual displays, transilluminated displays, scale indicators, audio displays, general controls, hand-operated controls, foot-operated controls, HCI, labeling, remote-handling environment, operation and maintenance, hazards and safety, physical access, protective clothing, and communications, as applicable and appropriate. The final design review is in accordance with the DOE orders listed in Section B13 2. The results are described in SNF-4213.

Human factors criteria are included in the system design descriptions for each SSC in the CVDF (SNF-2356). Each system design description includes a separate section that discusses the human actions performed while operating the system and the pertinent human factors issues associated with those actions.

The design requirements include HFE aspects at each step. For the purposes of developing requirements and ensuring their appropriate inclusions, the project engineers responsible for design are responsible for the applicable HFE requirements. This process is similar to promoting accountability downward to the lowest practical level. In addition, CVDF safety-class equipment RFPs include HFE requirements, as applicable. Finally, CVDF project personnel are aware of applicable HFE criteria, and their design reports and activities make reference to HFE considerations (see Figure B13-3).

B13 3 4 Designing for Off-Normal and Emergency Operations

An assessment team conducted a hazard analysis to identify potential hazards to the CVDF operator, co-located personnel, the public, and the environment (HNF-SD-SNF-HIE-004). Potential events considered ranged from normal operations to design basis accidents and natural events (e.g., seismic). The hazard analysis process consists of identifying the various operational steps to be conducted within the CVDF, and by doing so, considered HMI. The hazard analysis identifies both engineered and administrative safety features. Engineered safety features are design features that prevent or mitigate the potential accident and may require HMI. Administrative safety features are procedures and programs designed to ensure that engineered safety features continue to be available for accident prevention or mitigation.
safety features also include programs that implement a particular set of safety practices
SNF-4213 outlined possible areas of human error during a task analysis and recommended ways
to mitigate the potential human error, however, this HFE analysis was not a human reliability
study Chapter 3 of the SNF Project FSAR describes the hazard analysis methodology and
results for the CVDF are summarized in Chapter B3 0

In addition, the CVDF design team and Operations personnel reviewed off-normal
assessments, and reported via facilitated discussions (Ares 1998) on their hazard analysis of the
cold vacuum drying system. The team performed the off-normal assessment and identified and
assessed operational upsets (e.g., valve failures, control upsets, mechanical equipment failures)
that potentially could occur during normal operating conditions. The team examined each upset
condition to identify design and operational effects. The main objectives of this assessment were
to (1) identify and develop automatic monitoring and control system (MCS) actions to protect
against unacceptable operational consequences for the upset, (2) identify and develop
administrative protective actions, (3) identify any safety impacts that could result from the upset
or recovery from the upset, and (4) identify MCO “safe states” to be operationally achieved, if
necessary, from each step in the operations sequence. Ultimately, if an operational “safe state”
cannot be established or maintained, the safety-class instrumentation and control (SCIC) system
will trip to ensure that a “safe and stable” condition is provided. This effectively shuts down the
process in a safe condition until the original cause of the trip of the SCIC is determined and
remedied. The team is finishing work on specific pre-approved operational responses that will be
needed following an off-normal event to ensure that the facility safety envelope and boundaries
are not challenged.

The following information identifies what the operators at the CVDF are doing during
various stages of system processing (in addition, Section B13 3 1, “Process Background,”
provides additional detail of the HMI along with Table B13-1, as indicated above) Human
system interaction within the CVDF focuses mainly on two important areas: the process bay(s)
and the Central Control Room. The process bay provides direct human access to the cask-MCO
and exposes the human operator to a moderate risk. The control room provides the human
operator a means for surveillance, management, and control over the cask-MCO processing
operation(s). For purposes of the human factors safety basis, the control room is a non-reactor
control room.

The normal human operations at the CVDF are detailed in SNF-2356 Spent Nuclear Fuel
Cold Vacuum Drying Facility Operations Manual. These operations are outlined below

- Operators connect hoses to the tractor exhaust and hook up building air

- Operators raise the bay door to allow the tractor-trailer to back into the
  predetermined position. Location marks are painted on the floor to facilitate the
  trailer parking position. The tractor drives out of the bay, and the bay door closes
  achieving process bay confinement. The trailer is stabilized and leveled. The security
  system for the specific bay activates. The radiological control technician conducts...
### Table B13-1 Systems Reviewed for Human Factors Consideration (5 sheets)

<table>
<thead>
<tr>
<th>System reviewed</th>
<th>Significant HMI</th>
<th>HFE criteria compliance</th>
</tr>
</thead>
<tbody>
<tr>
<td>Safety-class instrumentation and control system</td>
<td>Control panels located in the control room provide an audible alarm to differentiate this system from the MCS. Responses to the SCIC annunciator are by operator action, the primary HMI is the MODE SWITCH, ANNUNCIATOR SWITCH, and the LOW ANNUlus LEVEL BYPASS SWITCHES. HMI is through test switches and reset functions. The SCIC system provides active detection and response to process anomalies. Specifically, actuation of the SCIC system performs two safety-class functions: (1) signal closure of isolation valves leading to and from the MCO to the process systems and signal opening the SCHel system isolation valves to provide MCO purging and pressurization, and (2) to remove power from the tempered water heater and pump motor. The SCIC system also provides safety significant annunciation to the control room if the PWC system prepurge or PWC system post purge process fails. The SCIC system is a fully automatic system that requires no operator action to detect and respond to process upsets within the SCIC system controls.</td>
<td>Yes meets applicable HFE general design criteria</td>
</tr>
<tr>
<td>Safety-class helium system</td>
<td>Check gauges and test/replace helium cylinders as required. The SCHel system is a dedicated safety system that actuates upon SCIC signal trip or is manually activated by a control room operator. The key function of the SCHel system is to pressurize and purge the MCO with sufficient helium pressure and flow to preclude flammable concentrations of hydrogen. A safety-class alarm is provided in the control room to signal that an actuation of the SCHel system has commenced.</td>
<td>Yes meets applicable HFE general design criteria</td>
</tr>
</tbody>
</table>
Table B13-1  Systems Reviewed for Human Factors Consideration  (5 sheets)

<table>
<thead>
<tr>
<th>System reviewed</th>
<th>Significant HMI</th>
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</tr>
</thead>
<tbody>
<tr>
<td>Safety-class annulus water protection system</td>
<td>Connect to tempered water system disconnect, flush</td>
<td>Not complete because long handled tools are being redesigned and will need HFE follow up</td>
</tr>
<tr>
<td></td>
<td>The operators using specially designed tools connect two water lines to the cask-MCO One water line is connected near the bottom of the cask, which serves as the tempered water inlet, and the other water line is connected in the cask-MCO seal ring which serves as the annulus outlet. The function of the tempered water (annulus) system is to maintain the MCO at the proper operating temperature during processing. The tempered water (annulus) system has three safety functions: (1) maintain water in the cask-MCO annulus (2) maintain cask-MCO inlet water temperature at less than 50°C (122°F) and (3) provide cask-MCO water level low level alarm and indication.</td>
<td></td>
</tr>
<tr>
<td>Cask-MCO</td>
<td>Inspect and decontaminate cask lid. Unbolt twelve 300 ft lb torque bolts, remove cask lid with overhead crane, install process hood ventilation, remove cask cover plate for tempered water, hook up tempered water hoses. Remove hood/seal ring. Install cask lid and torque 12 bolts by hand tool to 300 ft Ib ± 10%</td>
<td>Not complete because of HFE recommended design changes given to design authorities. Requires further HFE monitoring (as part of the Human Engineering Program Plan) to ensure that ergonomics (biomechanical) considerations related to special tools and torque wrenches are included.</td>
</tr>
<tr>
<td>MCO (process port)</td>
<td>Remove and install process connectors same with covers</td>
<td>Yes process port connectors and valve operators are spring tensioned to start, then manually torqued. Need to monitor installation of bolts with gloved hand.</td>
</tr>
<tr>
<td></td>
<td>The MCO is connected to vacuum purge system hardware via flexible hoses that run from the MCO process connectors to valves on the process hood support stand. This system is important to confinement.</td>
<td></td>
</tr>
</tbody>
</table>

Table B13-1: Systems Reviewed for Human Factors Consideration (5 sheets)
Table B13-1 Systems Reviewed for Human Factors Consideration (5 sheets)

<table>
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<th>System reviewed</th>
<th>Significant HMI</th>
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</tr>
</thead>
<tbody>
<tr>
<td>Process equipment and PWC skids</td>
<td>Routine maintenance as required Valve lineups and drain condensate Major maintenance if valve fails Electrical (instrument calibration) interfaces and mechanical interfaces within code compliance</td>
<td>Yes however accessibility areas (reading gauges levers filter changeout) need to be re verified with the selected vendor during acceptance tests</td>
</tr>
<tr>
<td>Overhead crane</td>
<td>HMI with control (eight control buttons) grappling hook has to be maneuvered from lid to hood to lid</td>
<td>Yes meets Hanford crane and ANSI crane specifications Will have eight control buttons in a standard configuration used on the Hanford Site</td>
</tr>
<tr>
<td>Heating ventilation, humidity control air conditioning lighting vibration, noise (environmental system)</td>
<td>Environment is designed using accepted national codes and standards There is no actual HMI except for routine maintenance and possible HEPA filter changes</td>
<td>Yes meets HFE applicable general design criteria and guidelines Maintenance accessibility and labeling are acceptable</td>
</tr>
</tbody>
</table>

The PWC will treat, sample and temporarily store process water drained from the MCOs in the CVDF. The PWC system provides confinement of contaminated effluents in the process water. Radiation protection from the effluents, worker safety hazards, protection criticality control, and natural phenomena hazard mitigation.

The HVAC will provide continuous water treatment and temporarily store process water drained from the MCOs in the CVDF. The PWC system provides confinement of contaminated effluents in the process water. Radiation protection from the effluents, worker safety hazards, protection criticality control, and natural phenomena hazard mitigation.

The HVAC has a computer that will include HCI Software is being developed and will include HCI considerations. The HVAC will need HFE followup.
**Table B13-1  Systems Reviewed for Human Factors Consideration (5 sheets)**

<table>
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<tr>
<th>System reviewed</th>
<th>Significant HMI</th>
<th>HFE criteria compliance</th>
</tr>
</thead>
<tbody>
<tr>
<td>CVDF structures</td>
<td>No HMI during operations except manually maneuvering bridge between mezzanine and cask-MCO work platform. There are 25 steps from the floor of the process bays to the mezzanine area. CVDF structure provides protection from seismic events, high winds, and tornado winds. The main structures of the CVDF provide the following functions: secondary confinement (except the administrative building); radiation protection (ALARA exposure from cask-MCO and primary process confinement of contaminated effluents); fire protection; natural phenomena and hazard mitigation (seismic, tornado, and high winds, floods) and worker safety hazards protection (industrial safety and industrial hygiene). Process bays are considered general service.</td>
<td>Meets HFE general design criteria and guidelines. All equipment requiring routine maintenance will have met accessibility requirements. Labeling meets applicable HFE guidelines. Bridging technical design — not known at this time how operator will accomplish this task.</td>
</tr>
<tr>
<td>Monitoring and control system</td>
<td>Significant HCI using computers, monitors, and mouse input devices. Complex screen presentations (object-oriented color displays). Response to both auditory and visual alarms as appropriate. Continuous monitoring activity. The MCS is the interface between the operators and the CVDF process systems. Its function is to provide active indication, alarm, and control of MCO processing and support functions throughout the facility. The MCS is classified as general service.</td>
<td>Software design considers HCI including NUREG-0700 guidelines. Software also considers IEEE standards for guidance and insight. Will meet applicable general HFE design guidelines. Meets applicable HFE checklist for software design considerations. Need to consider administrative procedures to reduce continuous monitoring by human.</td>
</tr>
<tr>
<td>Communication system</td>
<td>None except for use during normal and abnormal operations. The communications system provides operations personnel with the ability to communicate within the facility for efficient operation of the facility systems. The communication system interfaces with the 100 K Area communications the Hanford Local Area Network system, and the local telephone network. The CVDF communications system consists of a voice paging system, a standard telephone system, and a data communications system. While the communications system provides information during emergency situations of significance to personnel safety, the system has no direct safety function at the CVDF. The communication system is classified as general service.</td>
<td>Yes complies with applicable HFE criteria.</td>
</tr>
</tbody>
</table>
### Table B13-1 Systems Reviewed for Human Factors Consideration (5 sheets)

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</tr>
</thead>
<tbody>
<tr>
<td>Process water recovery</td>
<td>Limited HMI routine maintenance Periodic draining of tank</td>
<td>Yes but check test and validation phase for skid accessibility requirements</td>
</tr>
<tr>
<td>Central control room</td>
<td>Two operators use computers to monitor and control CVDF processes Video display terminal workstations The control room (room 107) has space for process bay control functions including computer monitoring stations MCS programmable logic controller cabinet space SCIC system annunciator and mode control panels security monitors (two) and egress alarm panel and HVAC control panels The control room measures 12 ft by 20 ft. The central control room is classified as general service</td>
<td>Yes Workstations are well designed with maximum adjustability Above average ergonomically designed chairs provided Potential glare on the computer screen concerns with lighting</td>
</tr>
</tbody>
</table>

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ALARA = as low as reasonably achievable
ANSI = American National Standards Institute
CVDF = Cold Vacuum Drying Facility
HCI = human–computer interface
HEPA = high-efficiency particulate air (filter)
HFE = human factors engineering
HMI = human–machine interactions
HVAC = heating, ventilation, and air conditioning
IEEE = Institute of Electrical and Electronics Engineers Inc
MCO = multi-cylinder overpack
MCS = monitoring and control system
PWC = process water conditioning
SCHe = safety-class helium
SCIC = safety-class instrumentation and control
SNF = spent nuclear fuel
radiation surveys on the cask and trailer CVDF shift operations management receives the shipping data package. The operators install a bridge from the process bay mezzanine to the trailer work platform. The operators prepare the cask lid for removal.

Special venting hardware and flex lines connect to the cask lid port and the CVDF process vent system. After cask venting is completed, the CVDF process bay overhead crane removes the cask lid, the process hood-seal ring is installed on the cask, process connectors are manually attached to MCO process port covers, and the MCO is prepared for process operations.

There are minimal manual operator actions in the process sequences other than field operator actions (e.g., connecting the MCO valves and flex lines) or control room operator actions (e.g., acknowledging alarms or instructing the MCS to proceed with the next step in the operation). Control room operators direct the MCS to initiate a certain sequence based on status updates from field operators and the MCS, previous sequence completion notification, and operating procedures. This includes draining the cask-MCO annulus.

Following the cold vacuum drying process and MCO testing, the cask-MCO transporter is prepared for shipment to the Canister Storage Building. This operation is basically the reverse of the receipt operation. Through operations controlled via the MCS, the cask-MCO is cooled and the MCO is inerted and pressurized with helium, sealed, and leak tested. The cask annulus is drained and dried with an instrument air purge, and the cask lid is reinstalled. The bay is isolated from the ventilation systems, and the telescoping door is opened. The trailer is reconnected to the tractor and released for shipment to the Canister Storage Building.

There are periodic nonintrusive routine maintenance requirements (the CVDF does not need shutting down) that require recalibration of instruments and possible high-efficiency particulate air (HEPA) filter changes. The probability of major failure with the resulting removing and replacing of major items, such as pumps, is extremely low. The systems are expected to last the entire time that SNF is being removed from the K Basins, which is expected to be 2 to 5 years.

Almost all human operator functional activities will be in the process bay, the PWC tank room (ion-exchange module changeout), and the Central Control Room. The tasks in the bay areas include connecting and disconnecting hoses, tightening applicable bolts, using tools identified in SNF-4213, manipulating the crane, fitting and unfitting the cask hood, and general “housekeeping” to keep the areas clean and safe.

The CVDF system is designed to respond to an incident and place the MCO in a safe configuration to ensure little opportunity for contaminated material release. Human interaction at this time will be to confirm that the MCO is in fact in a safe configuration, through the MCS and SCIC, and then take the appropriate actions per checklist, including fault isolation actions, as appropriate.
The HMIs for the engineering safety features were analyzed in accordance with the same criteria established in Section B13.2. Resolutions of noncompliance issues are documented in SNF-2825. In addition, SNF-4399 provides detailed information on how noncompliance issues are handled.

During the design of the CVDF, the architect-engineer used the hazard analysis or other analysis methods (e.g., failure mode and effects analyses) to evaluate the need for features or actions to increase safety and reliability. These analyses identified equipment items that were either included in, or removed from, the design of the CVDF to simplify the operations or improve reliability. The hazard analysis process also identified hazards resulting from identified maintenance activities. These hazards, if they result in accidents or contribute to an accident, will result in preventive or mitigative features or features that will decrease the frequency of the initial maintenance problem.

A job analysis was performed to identify the scope of the job and associated critical tasks. Based on the results of the job analysis, training is developed as described in Section 12.4 of the SNF Project FSAR. This process, along with design authority and management input, is the basis for determining the specialized maintenance capabilities required (e.g., vendor support or in-house journeyman knowledge). In addition, the SNF Project analyzed the types and complexity of equipment included in CVDF design and, applying past experience with these equipment types, identified the types and numbers of the various craftsmen needed for the maintenance support. During initial full operation of the CVDF, maintenance response teams will be assigned to each shift. The maintenance response teams will be composed of a cross-section of craft and technical support, including an engineer, a millwright, an electrician, and an instrument technician. This diversified team will have the capability to identify, diagnose, and repair problems with equipment that is essential to the facility function.

DOE Order 4330 4B specifies that maintenance facilities, equipment, and tools should efficiently support facility maintenance and maintenance training. The SNF Project has evaluated the maintenance needs in this area and the necessary office areas, tool and equipment storage, equipment access, and utility services layout (e.g., instrument air, water, electric power) to support safe and effective maintenance at the facility.

Measuring and test equipment includes all devices or systems used to inspect, test, calibrate, measure, or troubleshoot an instrument or piece of equipment to verify conformance to specified requirements. Only measuring and test equipment that is calibrated to standards that are traceable to the National Bureau of Standards or other nationally recognized standards are authorized for use by SNF Project personnel. Instruments are calibrated at specified intervals, before and after use, or just prior to use as determined by required accuracy, intended use, frequency of use, stability characteristics, and other conditions affecting performance.

There are no known limitations at the SNF Project facilities imposed by design and operation on routine maintenance, renewal, or repair of SSCs. Equipment in hazardous and radiological areas is designed to be free of maintenance as much as possible and is operated in a...
“run-to-failure” mode. Accessibility (physical access) for maintenance and provisions to support effective maintenance were considered in the facility design.

B13 4 IDENTIFICATION OF HUMAN–MACHINE INTERFACES

This section summarizes the safety-class and safety-significant SSCs that require HMI to function and the associated HMI. Also annotated are the significant HMIs that are pertinent to the operation of the CVDF. The intent during design was to reduce, as much as possible, HMI at the CVDF. The system is designed to run very efficiently in an automated state, however, human input (e.g., providing permission for the system to move from one process to the next process) assures that the system is maintained under human control through all processes and keeps the human involved at critical stages of the process operation. The human will also monitor system operation. When there is an off-normal occurrence, the human physically checks the discrepant part of the system. Periodic human inspection of the system and routine periodic maintenance will also be required.

Using the DOE-recommended graded approach and DOE Order 6430.1A, the analyst performs a systems requirements analysis as an integral part of the design. Therefore, the systems approach to design is applied whereby systems, structures, and equipment are built in accordance with the best available design practices. HNF-SD-SNF-DRD-002, Cold Vacuum Drying Facility Design Requirements, includes a number of human factors considerations in the CVDF design. The application of these human factors interface guidelines to the CVDF is documented for the safety-related SSCs. Table B13-1 shows the systems that were reviewed for human factors considerations, additional detail concerning HFE analyses of these systems is provided in SNF-4213. SNF-4399 describes human engineering processes to be used in design analyses. Subsequent analyses are reported in SNF-4213.

B13 5 OPTIMIZATION OF HUMAN–MACHINE INTERFACES

In considering the optimization of HMI, the reduction of human involvement to the minimum levels necessary to ensure complete facility safety stands out as a premier design concept. The staffing levels, training, special tools required, HCl, and response to off-normal conditions consider human interfaces. Optimized alarms allow the human operators to quickly and accurately identify the condition generating the alarm and to determine the next course of action.

Tools are being selected with consideration of optimizing human performance. Optimization does involve a cost-benefit trade-off and one such area is the trade-off involving the 300 ft-lb bolt torquing requirements for the cask lid. Machinery would have been a logical choice, however, a technique using a torque multiplier on the manual torque wrenches is appropriate and more cost effective under these circumstances. The multiplier will have a six-to-one ratio designed to reduce the actual human-provided push and pulling forces to manageable levels.
Layout and design of controls, instruments, and provisions for labeling that apply the principles of ergonomics and human engineering were evaluated using the applicable INEL-95/0117 checklists and applying those checklists to a mock-up of the CVDF bay work platform surrounding the cask (as if the trailer-cask was parked in the bay). Also, the support skids were mocked up and evaluated, along with simulation of the control room with computer displays for monitoring and controlling the CVDF operation. With input from the checklists and interviewing operators, the system provides for effective HMI design.

Consideration is given to minimum staffing levels, training, and periodic maintenance, these considerations are described in Section B13. As stated above, the premier design concept was to minimize allocation of control functions to humans, creating automated processes where appropriate. However, the human component retains control over the entire process because the process will stop after appropriate sequences and wait for the human to initiate the next required sequence.

The work environments meet general operating HFE environmental design criteria, based on available calculated engineering design requirements and comparing the results to existing criteria found in the HFE standards and checklist. For example, calculated room lighting in the control room appears to meet acceptable standards. Physical access was evaluated using the mockup and appears acceptable. Also, input from the operators was solicited and provided to the skid vendors. Local Hanford Site procedures and requirements provide personal protective clothing requirements.

Information about temperature, humidity, and other environmental conditions is provided in Chapter B2.0 and in the system design description for the heating, ventilation, and air conditioning system. The available calculated values and available design information from system design descriptions were compared to DOE-STD-1062-94, as captured in INEL-95/0117. The HFE checklist examines temperature, humidity, ventilation, noise, illumination and other factors impacting human operations during normal operations, including routine periodic maintenance. SNF-2825 and SNF-4213 provide an examination of the facility environment.

Software design is being optimized by using the required engineering standards and guidelines, and includes operator input via the mock-up and software designer experience. These experiences and mock-up lessons learned contribute to the software development, making the operation of the CVDF intuitive and easy to follow for the operator(s). The system software manages the physical computer resources on behalf of the application software. The MCS is the interface between the control room operator and the process system. It provides process values and status in graphical (mimic) formats to the operator via a computer screen. The MCS accepts operator commands and overrides.
Functionally, the MCS monitors the instrumentation, controls equipment, displays status information, and accepts operator input. The MCS design and program provide the following process functions:

- Monitor and display the status and performance of each selected cold vacuum drying treatment process subsystem and component group.
- Provide a means to remotely control selected process control loops by the operator during operation (e.g., adjust the tempered water operation setpoint).
- Provide audible and visual alarms for out-of-limit conditions on various process parameters and facility systems.
- Provide automated computer control of system processes and provide the means to take manual control (operated from the computer console) of a discrete process component.
- Provide process control, interlock, and alarm logic as programmed into the programmable logic controller, which provides for all proportional/integral/derivative loop controls.
- Provide for storage of selected data and method(s) for archiving data.
- Provide for retrieval, display, and printout of historical data, real time data, and selected trends.
- Provide a prioritized and time/date-stamped alarm summary on the MCS display, alarm status printouts are available on demand only.

The CVDF sequences are automatic, once initiated. There will be points in the sequence where the operator or supervisor will be required to interact with the control system. Adjusting a setpoint and other control parameters requires operator input. The control system will maintain operator-entered values.

The MCS software design considers the use by process operators. All computer displays and windows are representative of the operating sequence, the process flow, and components. All persons who operate this equipment will receive training on the MCS and its use.

Visual displays and annunciator panels include coding and display integration methods in accordance with DOE Order 6430 1A, Section 1300-12 4 7, and its references that include applicable sections of both NUREG-0700 and MIL-STD-1472E, *Human Engineering*. Technical HCI work is evident and has received special attention by the design engineers.
The MCS consists of two operator workstations, an engineering workstation, a redundant programmable logic controller, and locally mounted input/output modules with termination panels and interfacing relays. The operator interfaces are object-oriented color graphic displays on computer screens located in the central control room and management office. An open systems network connects displays but can only be accessed (changed) in the central control room by the operators. This allows simultaneous access to the control system from multiple graphic displays. The MCS is proprietary and there is full control from any of the stations in the control room. The supervisory computer, located in the supervisor’s control room, is read-only.

The MCS boundary includes all components starting at the workstations and ending at the wiring terminations at the process bay skids. The actual sensors, valves, and other process equipment are outside the MCS boundary. The MCS interfaces with the following processes in each of the four bays (2 through 5):

- Helium supply system
- Tempered water system
- Vacuum pumping system
- De-ionized water system
- Process water conditioning

Also, the MCS interfaces with:

- Radiation monitoring
- Stack monitoring

Checklists support the optimization process of integrating the human into the system design. An HFE checklist is similar to a questionnaire. It contains a list of items pertaining to the human engineering aspects of equipment. One or more evaluators can rate the adequacy of the equipment on each applicable item of the checklist. Checklists are most useful in the early phases of systems development and design where concern is primarily with static human engineering design properties. For example, the placement of displays and controls for adequate viewing, the labeling of instruments and equipment, and other features not directly related to human-machine performance. Checklists have limited usefulness in the evaluation of dynamic human interaction with equipment. Measurements of this kind need to be observed and recorded directly. The checklists used for CVDF HFE and ergonomics evaluation are constructed from several standards that typically evaluate the design based on 5th to 95th percentile personnel population expected to operate and maintain the facility. Design optimization for this range of personnel generally has been considered as optimum based on the cost of the design, the time to design, and other factors (e.g., materials available at the time of design). Checklists developed from HFE standards ensure the design will meet minimum HFE criteria based on best available information and consensus.
Normal human operations were reviewed and reported in a human factors report, SNF-2825, and a follow-up human factors report, SNF-4213. These human factors study reports document the tabletop task analysis, systems reviews, direct interviews with design authorities and cognizant engineers, and reviews of applicable design documentation. The INEL-95/0117 checklists are used to supplement the task analyses, interviews, and design documentation reviews. In addition, the task analysis done by the training professional staff was captured in the draft procedures training document, which also was reviewed. The training document provides the step-by-step procedure that an operator will accomplish during a task activity. These procedures will be validated during the dry run test and evaluation phase of the project, as described in SNF-4399.

The normal human operations at the CVDF are detailed in SNF-2356. These operations are outlined as follows:

- Operators connect hoses to the tractor exhaust

- Operators raise the bay door to allow the transporter to back into the predetermined position. Location marks are painted on the floor to facilitate the transporter parking position. The tractor drives out of the bay and the bay door closes, achieving process bay confinement. The transporter is stabilized and leveled. The security system for the specific bay activates. The radiological control technician conducts radiation surveys on the cask and transporter. CVDF shift operations management receives the shipping data package. The operators install a bridge from the process bay mezzanine to the transporter work platform. The operators prepare the cask lid for removal.

- Special venting hardware and flex lines connect to the cask lid port and the CVDF process vent system. After cask venting is completed, the CVDF process bay overhead crane removes the cask lid, the process hood-seal ring is installed on the cask, process connectors are manually attached to MCO process ports, and the MCO is prepared for process operations.

- There are minimal manual operator actions in the process sequences other than field operator actions (e.g., connecting the MCO valves and flex lines) or control room operator actions (e.g., acknowledging alarms or instructing the MCS to proceed with the next step in the operation). Control room operators direct the MCS to initiate a certain sequence based on status updates from field operators and the MCS, previous sequence completion notification, and operating procedures. This includes draining the cask–MCO annulus.

- Following the cold vacuum drying process and MCO testing, the cask–MCO transporter is prepared for shipment to the CSB. This operation is basically the
reverse of the receipt operation. Through operations controlled via the MCS, the cask-MCO is cooled and the MCO is inerted and pressurized with helium, sealed, and leak tested. The cask annulus is drained and dried with an instrument air purge and the cask lid is reinstalled. The bay is isolated from the ventilation systems, and the telescoping door is opened. The transporter is reconnected to the tractor and released for shipment to the CSB.

There are periodic nonintrusive routine maintenance requirements (the CVDF does not need shutting down) that require recalibration of instruments and possible high-efficiency particulate air filter changes. The probability of a major failure with the resulting removal and replacement of major items (e.g., pumps) is extremely low. The systems are expected to last the entire life cycle of the facility, which is stated as five years.

Almost all human operator functional activities will be in the process bay, the process water conditioning tank room (ion-exchange module changeout), and the central control room.

B13 5 2 Staffing and Training

B13 5 2 1 Operator Capabilities. Skills, knowledge, and abilities to meet job standards are defined to ensure that personnel have the appropriate capabilities to perform the required activities in a safe and reliable manner. As stated in Chapter 17 of the SNF Project FSAR, minimum education and experience requirements for management and technical personnel meet the requirements of DOE Order 5480.20A Qualification criteria define the experience, education, and training required to perform a designated job. Chapter 17 of the SNF Project FSAR and Chapter B5 also provide a discussion of the minimum staffing requirements.

Verification of operator capabilities occurs throughout the testing and validation review programs. Specifically, the operational readiness reviews will include dry run demonstrations that ensure operators and procedures are in a satisfactory state of readiness to safely and efficiently receive, handle, and process MCOs. The dry run demonstrations permit certification of both procedures and the operations staff by confirming that personnel have the appropriate knowledge and capabilities to perform their duties.

The completed safety analysis report process for selecting safety-class and safety-significant SSCs and technical safety requirements also serves as a systematic inquiry into the ability of facility staff to accomplish their responsibilities and duties during normal and abnormal operations.

B13 5 2 2 Staffing Levels. A current staffing plan exists for the CVDF through 2002. A review of this plan indicates that appropriate consideration was given to the following:

- Learning curves
- Loss of trainees through the process
- Appropriate job titles and functions for operations personnel
One factor that appears to weigh heavily in the operations staffing requirement concerns the occupational radiation exposure calculations. For the CVDF, occupational radiation exposures to workers are estimated based on the manual operations performed in the presence of the radiation field surrounding one cask-MCO in the CVDF. As discussed in Section B 7.10, the planned staffing level supports the annual MCO processing goal (147 MCOs) with a resulting annual exposure to the working staff well below the ALARA goal set by HSRCM-1, *Hanford Site Radiological Control Manual*.

Consideration for retaining a three-shift work day concerns the commute to and from the CVDF. Operators at the MCS control station may be mentally fatigued from constant surveillance and monitoring. Having operators work longer hours under these conditions may impose a greater safety risk to the operator when commuting from work to home while mentally fatigued. The commute will most likely be by privately owned vehicle.

**B13 5 2 3 Personnel Training** Training ensures that all staff have the knowledge, skills, and abilities required to safely operate and maintain the facilities and processes. As discussed in Chapter 12.0 of the SNF Project FSAR, the SNF Project training organization applies DOE-mandated performance-based training to specific job requirements. This is being implemented using a systematic approach to training development that is in compliance with the requirements of DOE Order 5480.20A.

The SNF Project uses HNF-PRO-170, *Analyzing Training Requirements*, HNF-PRO-171, *Designing Training*, and HNF-PRO-174, *Evaluating Training*, as guidelines in developing and implementing an effective training program. As identified in Section 12.4 of the SNF Project FSAR, this systematic approach to training ensures that the training program will have the following characteristics:

- Training based on a systematic analysis of each CVDF job position
- Measurable learning objectives derived from the systematic analysis
- Desired post-training performance described
- Trainee mastery of objectives evaluated during training
- Job performance of trainees used as a basis for evaluation and revision of training

Operators receive on-the-job training that consists of learning to operate the specific components, equipment, and systems in the CVDF. The majority of on-the-job training package development and operator evaluations will be performed during pre-operational testing and validation. Pre-operational testing and validation includes dress rehearsals using mock-ups and in situ equipment and systems to provide verification of the operators' knowledge of facility procedures, readiness of tools, equipment, and instrumentation, and off-normal and emergency procedures. The major focus is incorporating job and task analysis information into the on-the-job training material and ensuring that the operators are trained and evaluated on this material.

Schedules, material, and evaluations for continuing training are being developed. These include (1) classroom training on fundamentals, systems, administrative controls, and emergency
response, and (2) on-the-job training for normal and abnormal operation. Job and task analyses also assist in defining those tasks that require continuing training. To ensure that training properly reflects the current operating practices, conditions, and procedures, a training materials maintenance system is used to track items that may affect the content of the training programs. Chapter 12.0 of the SNF Project FSAR provides additional details on the maintenance of training programs and the use of a training materials maintenance system.

Operational drills and exercises are an integral part of the facility training program. Drills are an effective means of providing training in the handling of off-normal conditions and situations. Drills also provide management with an assessment tool to help evaluate the staff's ability to respond to off-normal events and conditions. Drills and exercises provide the means for testing crew proficiency and validating procedures.

Task statements and task elements within the scope of the training task analysis are complete for the CVDF first article process equipment skid. A student training manual is being developed from the training task analysis.

B13 5 2 4 Procedures DOE-STD-1029-92 states, "The overall safe operation of the facility depends upon the structured interrelationship among DOE requirements and guidance, the bases documentation (senior management, technical, management control, and design) and the facility’s procedures — generally referred to as its safety envelope.” In other words, a facility’s procedures define how requirements, management philosophies and strategies, and technical knowledge are integrated and applied to performing work in the facility. See Section 12 3 of the SNF Project FSAR for a complete discussion of the SNF Project procedures program.

DOE-STD-1029-92 was created by the Human Factors Area Council to ensure writers of technical documents will follow basic human factors principles in their content, layout, and format of written technical procedures. Because DOE-STD-1029-92 complies with accepted human factors principles, implementation of this standard demonstrates that human factors considerations are included.

In addition to training or day-to-day supervision, providing sound procedures and requiring workers to use them are among the most formal, direct, and effective methods available to facility managers to ensure that operations meet DOE’s objectives. Procedures provide managers with a critical tool to communicate detailed expectations for how individual workers are to perform specific tasks. In other words, the facility’s operating procedures provide the mechanism for ensuring that facility operation is maintained within the safety basis.

During the validation portion of the review process for draft procedures, the procedure will be tested for correctness, compatibility with system equipment, and human factors considerations and usability. Validation includes determining whether the user can perform each step correctly and ensuring the units used in the procedure are consistent with the facility instrumentation and possess the required precision for operation. Validation ensures the sequence of steps is logical and can actually be performed, all documentation (e.g., labels, equipment representations, and
diagrams) contained in the procedure is accurate, and tasks required by the procedure can be effectively performed by operations personnel during normal, off-normal, and emergency conditions. Alarm response procedures are developed on the basis of the monitoring and control functions of the system or component providing the alarm output. All operating procedures are validated by procedure walk-through and simulated operation before they are used. End-user or operator participation in both the procedure writing and the review processes is solicited.

Procedures will be periodically reviewed. This periodic review process ensures the technical accuracy and the proper considerations of human factors issues in procedures. Included in the review process are CVDF operating procedures, alarm response procedures, and surveillance and test procedures. The process involves verification of the technical basis and procedural steps, validation of the usability of the procedure by operations personnel, and verification of human factors elements that may affect the effectiveness of the procedure.

### B13 5 3 Abnormal Operations

Scenarios for abnormal operations involving human recovery, initiator, or mitigator actions are being completed (see Section B3 3 4). Typically, basic recovery response by operators is to acknowledge the abnormal occurrence and investigate the cause. The MCS and SCIC will automatically take the applicable action and wait for operator acknowledgment before restarting the procedure. Abnormal operations are fully covered in training and validation during pre-operational testing.

The minimum operations shift complement in the facility modes (i.e., normal operations, abnormal operations, and emergency operations) is established and based on the minimum staff in each mode considered adequate to perform the minimum safety functions necessary to protect the health and safety of the public, onsite workers, and the environment during normal, abnormal, and emergency conditions. The minimum staff during abnormal conditions is necessary to perform the required actions specified in the limiting conditions for operation action statements with completion times of "immediately" or less than eight hours to ensure technical safety requirement compliance. Hanford Site experience has shown that additional staff could be provided within eight hours, if needed (considering the most adverse weather and travel conditions) to ensure all limiting conditions for operation completion times are met. Limiting conditions for operation completion times of "immediately" imply the highest sense of urgency and are given top priority over all other activities. The minimum staff is not given tasks that could interfere with meeting technical safety requirements.

The minimum staff during emergency conditions is necessary to respond to the spectrum of accidents analyzed in Chapter B3 0. The minimum staff must make prompt initial notifications and implement initial protective actions to preclude or reduce the exposure of individuals affected by hazards or unsafe conditions during an emergency. Specific functions that must be performed by the minimum staff in an emergency include the following:
Implemention of alarm response, plant response, and emergency management procedures

- Classification of events
- Initial prompt notifications
- Performance of administrative functions (e.g., preparing occurrence reports)
- Communication of facility status and response to questions
- Support for the DOE, Richland Operations Office Emergency Operations Center

An emergency response plan is developed to be compatible and integrated with the disaster, fire, and/or emergency response plans of local, state, and federal agencies. The emergency response plan shall be rehearsed regularly as part of the overall training program for site operations. An employee alarm system is installed to notify employees of an emergency situation, to stop work activities if necessary, to lower background noise in order to speed communication, and to begin emergency procedures.

A spectrum of potential accidents ranging from minor to beyond design basis are postulated and realistically analyzed. While not every conceivable situation will be analyzed, the hazards assessments provide the framework for response to virtually any declared emergency. The methodology, assumptions, models, and evaluation techniques used in the hazard assessments are documented.

Operations response emergency equipment is stored in designated SNF Project locations. This equipment is used in all types of emergency events and includes Emergency Response Organization staff identification vests, emergency response procedures, personal protective equipment, spill kits, flashlight, barricade rope, and first aid kits.

The SNF Project emergency response training program ensures the readiness of the SNF Project emergency preparedness organization, and general employee awareness of hazards and response actions. This program is part of the overall Hanford Site training program and includes classroom instruction, tabletop exercises, walkthroughs, on-the-job training, and drills. Drills and exercises for this program are of sufficient scope and detail to ensure demonstration of adequate response capability in all areas and by all populations.

B13 6 REFERENCES


DOE Order 5480 19, Conduct of Operations Requirements for DOE Facilities, U S Department of Energy, Washington, D C

DOE Order 5480 20A, Personnel Selection Qualification and Training Requirements for DOE Nuclear Facilities, U S Department of Energy, Washington, D C

DOE Order 5480 23, Nuclear Safety Analysis Reports, U S Department of Energy, Washington, D C

DOE Order 6430 1A, General Design Criteria, U S Department of Energy, Washington, D C


HNF-PRO-170, Analyzing Training Requirements, Fluor Daniel Hanford, Incorporated, Richland, Washington

HNF-PRO-171, Designing Training, Fluor Daniel Hanford, Incorporated, Richland, Washington


Annex B — Cold Vacuum Drying Facility


Figure B13-1  Cold Vacuum Drying Facility Process Overview
Figure B13-2  Human Factors Process Integration with Total Design Effort
(Sheet 1 of 2)
Figure B13-2 Human Factors Process Integration with Total Design Effort
(Sheet 2 of 2)

Develop full scale mock-up of proposed designs skids, cask, MCS, etc

Examine human actions in mock-up environment and HCI

Provide CVDF cognizant engineer reviews at design stages

Complete HFE checklists against human engineering design criteria

Resolve human engineering deficiencies as revealed during checklist reviews

Further evaluate human interfaces

Provide continuing HFE review of design changes and incorporate solutions

Includes HFE criteria

Review programmatic concerns

Preoperational start-up

Verify and validate the entire HFE effort

Report final results of the HFE evaluation process

MCS = monitoring and control system
HFE = human factors engineering
HCI = human-computer interface
Figure B13-3 Human Factors Checklist Procedure

- Familiarization with CVDF systems, structures, and components
- Review the Facility Hazards Evaluation
- Select HMI to evaluate based upon those systems that are either Safety Class or pertinent
- Tabletop task analysis to determine task performance, sequence, and maintenance
- Interview cognizant project personnel to determine level of design and design efforts
- Review design requirements process, pertinent subcontract RFPs, and facility design process for inclusion of HFE criteria
- Application of HFE checklists to existing SSCs and design requirements documents
- Checklist evaluation results sorted into one of four categories: 1) complied, 2) not complied, 3) not applicable, and 4) to be determined
- Resolve any non-compliances
- Generate commitments to consider TBDs in continuing design efforts
- Review programmatic issues (staffing, maintenance, procedures, and training)
- Resolve any programmatic concerns
- Report on the results of the HFE evaluation
CHAPTER B14 0
QUALITY ASSURANCE
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November 1999
**LIST OF TERMS**

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<th>Description</th>
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<tr>
<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
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<td>DOE</td>
<td>U.S. Department of Energy</td>
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<tr>
<td>DOE-RL</td>
<td>U.S. Department of Energy, Richland Operations Office</td>
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<tr>
<td>FSAR</td>
<td>Final Safety Analysis Report</td>
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<tr>
<td>MCO</td>
<td>Multi-Canister Overpack</td>
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<td>MCS</td>
<td>Monitoring and Control System</td>
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<tr>
<td>NRC</td>
<td>U.S. Nuclear Regulatory Commission</td>
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<tr>
<td>QARD</td>
<td>Quality Assurance Requirements and Description</td>
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<td>SNF</td>
<td>Spent Nuclear Fuel</td>
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B14 0 QUALITY ASSURANCE

B14 1 INTRODUCTION

An introduction to the quality assurance program that includes the objectives and scope that apply to all Spent Nuclear Fuel (SNF) Project quality assurance activities is provided in Chapter 14 of the SNF Project Final Safety Analysis Report (FSAR).

B14 2 REQUIREMENTS

The requirements that form the basis of the quality assurance program are identified in Section 14 of the SNF Project FSAR. Additional requirements for the Cold Vacuum Drying Facility (CVDF) are identified in the following paragraphs. These requirements ensure the quality assurance requirements for federal repository acceptance of SNF and U.S. Nuclear Regulatory Commission (NRC) equivalency requirements are satisfied.

The U.S. Department of Energy, Richland Operations Office (DOE-RL), has directed (Sellers 1995) that DOE/RW-0333P, Quality Assurance Requirements and Description (QARD), published by the Office of Civilian Radioactive Waste Management, be applied as the principal quality assurance document to the SNF Project office of Civilian Radioactive Waste Management program. DOE-RL has directed application of DOE/RW-0333P to the following SNF Project activities as they relate to disposal in the repository:

- Characterization or data collection for input and use
- Conditioning into final form
- Handling, packaging, and transportation

Items activities and documentation determined to be important to safety are presented in Table 3-1 in HNF-SD-SNF-RPT-007, Application of the Office of Civilian Radioactive Waste Management Quality Assurance Requirements to the Hanford Spent Nuclear Fuel Project. HNF-SD-SNF-RPT-007 identifies certain high-level structures, systems, components, activities, and documentation that require application of DOE/RW-0333P requirements to ensure compliance with the DOE-RL direction. Application at the CVDF includes, but is not limited to, the following:

Items

- Equipment that ensures adequacy of water removal from a multi-camster overpack (MCO) at the CVDF and ensures that site safety bases requirements are satisfied for limiting particulate generation
Activities and Documentation

- Process validation tests and operational tests where necessary to demonstrate adequacy of process conditions for removal of water from the MCO and for limiting particulate generation

- Equipment acquisition and acceptance (e.g., specification, calibration) where necessary to ensure adequacy of water removal from the MCO and ensure that site safety bases are satisfied for limiting particulate generation

- CVDF process design specifications, design, design analyses, and design verification where necessary to ensure adequacy of water removal from the MCO and ensure that site safety bases limits are satisfied for particulate generation

- Measures to ensure unapproved materials are not introduced into the MCOs

- Data developed by CVDF personnel that are also needed by the U.S. Department of Energy (DOE) Office of Civilian Radioactive Waste Management, based on formally transmitted data requirements from DOE-RL

In addition, HNF-SD-SNF-RPT-007, Appendix B, identifies, at a lower level, specific items, activities, and documentation that require application of QARD requirements, including the following

**Items requiring application of QARD requirements**

**Components**

- **PT 1*08, PT 1*10**  
  MCO pressure transmitter (vacuum 0-100 torr)

- **TIT 3*05, 3*12**  
  Tempered water temperature, inlet and outlet

- **CP 210, 311, 410, 511**  
  Safety-class instrumentation and control system programmable logic controller panels with programmable logic controller analog card for PT and TIT signals (1 card each panel)

- **CP 201, 301, 401, 501**  
  Monitoring and control system (MCS) input/output panels with analog card for PT and TIT signals (2 cards each panel, modules 3 and 6)
Loops

- MCO pressure: Signal from pressure element to MCS printer output
- Tempered water temperature: Signal from temperature element to MCS printer output

Software

- MCS computer program: Plot routine and analog conversion of PT and TIT signals

Activities and items requiring application of QARD requirements

- Analysis and characterization test results showing the correlation of pressure and temperature to water quantity in the MCO
- Analysis (HNF-SD-W441-CN-001) to show corrosion generation will not exceed safety basis limits
- Design specification and design necessary to ensure adequacy of water removal and that particulate generation will not exceed site safety basis requirements
- Validation plan for CVDF water removal strategy, test procedures, and test results
- Selected operating procedures for CVDF drying
- Operating procedures for helium hookup, receipt inspection records on helium purity, and helium purity sample data
- Operating data records from QARD components
- Calibration records for QARD components
- Training records for operations and maintenance personnel involved with processes and components associated with QARD requirements

Data collected and controlled during the cold vacuum drying process to show compliance with the QARD requirements will be summarized and inserted into each MCO data package (traveler) prior to shipping the respective MCO to the Canister Storage Building, consistent with QARD requirements.
DOE has established a regulatory policy (Grumbly 1995) that new SNF Project facilities achieve nuclear safety equivalency with NRC-licensed facilities. An evaluation, documented in WHC-SD-SNF-DB-002, *Spent Nuclear Fuel Project Path Forward Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities*, identified requirements to establish nuclear safety equivalency that are to be met in addition to existing and applicable DOE requirements. These requirements, except those related to the design basis earthquake, are contained in HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward Additional NRC Requirements* HNF-SD-SNF-DB-004 *SNF Project Seismic Design Criteria — NRC Equivalency Evaluation Report*, contains the design basis earthquake requirements.

NRC nuclear safety equivalency requirements identified in HNF-SD-SNF-DB-003 that will be applied to the CVDF include:

- Incorporation of requirements into safety-class procurement specifications that requires suppliers to report defects and noncompliances in items or services (Item 15)
- Review and approval by DOE-RL of changes to HNF-MP-599, *Project Hanford Quality Assurance Program Description*, that could be interpreted as decreasing the quality assurance program’s existing commitments for the CVDF (Item 16)
- Implementation of the Project Hanford Management Contract occurrence reporting system for the design and construction of the CVDF (Item 17)
- Ensure the appropriate quality requirements in existing Project Hanford Management Contract procedures and instructions identified in WHC-SD-SNF-DB-002 remain in effect (Item 18)
- Institute a process to identify safety-class equipment that has been identified in the commercial nuclear power industry as being potentially defective (Item 19)
- Ensure the areas of vendor and subcontractor quality assurance records and control of purchased material, equipment and services considered important to safety receive emphasis during CVDF audits, surveillances, and assessments (Item 19)

Additional requirements apply to an MCO when it resides in the CVDF. These requirements are identified in HNF-SD-SNF-DB-005, *Spent Nuclear Fuel Project Multi-Canister Overpack Additional NRC Requirements*. Design requirements for natural phenomena hazards, other than seismic design requirements, are identified in WHC-SD-SNF-DB-010, *Cold Vacuum Drying System Natural Phenomena Hazards*.

The documents cited in this chapter identify the requirements to achieve nuclear safety equivalency with NRC-licensed facilities and to meet the requirements of DOE/RW-0333P. The quality assurance program plan for SNF Project facilities provides for implementation of these requirements. A graded approach will be used for items and activities important to safety.
B14.3 QUALITY ASSURANCE PROGRAM ORGANIZATION

A summary of the SNF Project quality assurance program, including summaries of safety management policies and philosophies used as a basis for the program, is provided in Section 14.3 of the SNF Project FSAR.

The SNF Project organizational structure, responsibilities, authorities, and interfaces that apply to the CVDF are addressed in Chapter 17.0 of the SNF Project FSAR.

B14.4 QUALITY IMPROVEMENT

Descriptions of SNF Project management programs and processes used to correct adverse conditions affecting quality at all SNF Project facilities are provided in Section 14.4 of the SNF Project FSAR.

B14.5 DOCUMENTS AND RECORDS

A description of the SNF Project document control and records management program associated with quality assurance is provided in Section 14.5 of the SNF Project FSAR.

B14.6 QUALITY ASSURANCE PERFORMANCE

An overview of the SNF Project process to ensure that the performed work meets requirements is provided in Section 14.6 and subsections in the SNF Project FSAR. The subsections address work processes, design activities, the procurement process, program tests and inspections, management assessments, and independent assessments.

B14.7 REFERENCES


CHAPTER B15 0

EMERGENCY PREPAREDNESS PROGRAM
Annex B — Cold Vacuum Drying Facility

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B15-1 Cold Vacuum Drying Facility Emergency Equipment
LIST OF TERMS

BED  building emergency director
BEP  building emergency plan
CVDF  Cold Vacuum Drying Facility
DOE  U S Department of Energy
EPZ  emergency planning zone
ERG  Emergency Response Organization
ERPG  Emergency Response Planning Guideline
FSAR  final safety analysis report
IC  incident commander
ICP  Incident Command Post
ONC  Occurrence Notification Center
PAG  protective action guide
POC  Patrol Operations Center
SNF  spent nuclear fuel
B15 0 EMERGENCY PREPAREDNESS PROGRAM

B15 1 INTRODUCTION

A description of the generic philosophy, objectives, and organization of the Spent Nuclear Fuel (SNF) Project emergency preparedness program for response to emergencies at the SNF Project facilities is provided in Chapter 15 0 of the SNF Project Final Safety Analysis Report (FSAR). This Annex B chapter presents emergency management information specific to the Cold Vacuum Drying Facility (CVDF).

B15 2 REQUIREMENTS

The requirements that form the basis for the SNF Project emergency preparedness program are identified in Section 15 2 of the SNF Project FSAR.

B15 3 SCOPE OF EMERGENCY PREPAREDNESS

Potential CVDF emergencies could span the spectrum of identified emergencies for SNF Project facilities from worker injuries to general emergencies with potential public impact. The spectrum of emergencies that the CVDF emergency preparedness program is designed to encompass is described in Section 15 3 of the SNF Project FSAR.

B15 4 EMERGENCY PREPAREDNESS PLANNING

SNF Project emergency preparedness planning includes identification of emergency organizations, assessment actions, notification processes, emergency facilities and equipment, protective actions, access control, training, drills, exercises, and recovery actions. A summary of the emergency response organization that is activated during CVDF emergencies is provided in Section 15 4 of the SNF Project FSAR and its subsections.

B15 4 1 Emergency Response Organization

Section 15 4 1 of the SNF Project FSAR presents information related to the organizational structure and support resources to meet emergency event requirements. This organization structure and support resources are applicable to the CVDF. The Hanford Site Emergency Response Organization (ERO) has the responsibility to take actions in response to a CVDF emergency to prevent or minimize impacts to workers, the public, facilities and the environment. Initial direction and control of emergency response at the CVDF, prior to establishment of an...
Incident Command Organization as described below, is the responsibility of the CVDF ERO. Key Hanford Site ERO positions and responsibilities are discussed below.

**B15 4 1 1 Organization Structure**

The Hanford Site ERO, responding to a CVDF emergency, will consist of two components, an Incident Command Organization and a U.S. Department of Energy (DOE) Hanford Emergency Operations Center as stated in Section 15 4 1 1 of the SNF Project FSAR.

**B15 4 1 2 Incident Command Organization** The Incident Command Organization will consist of the CVDF ERO and site contractor emergency personnel (i.e., Hanford Fire Department and Hanford Patrol). The Incident Commander (IC) is the senior Hanford Fire Department official, the Hanford Patrol will provide the IC for security emergencies. The IC will direct all emergency response efforts at the CVDF and has overall responsibility for the health and safety of all personnel at the event scene.

The CVDF ERO, including the BED, will become part of a consolidated Incident Command Organization and function under the direction of the Incident Commander. In this role, the BED retains responsibility for CVDF operations and direct configuration control over CVDF systems and components, the IC controls the overall management strategy associated with the emergency event and ensures that all functional areas are staffed and working effectively to mitigate the accident.

The Incident Command Organization has the authority to commit all SNF Project resources (equipment and personnel) in response to any emergency and to request supporting resources. Other responsibilities include (1) implementing the CVDF building emergency plan, (2) assuring that the CVDF ERO is fully staffed and trained, (3) initially assessing, categorizing and classifying events, (4) notifying the Patrol Operations Center and applicable contractor and DOE management through the Occurrence Notification Center, (5) implementing protective actions, (6) establishing an initial CVDF Incident Command Post, (7) controlling the event scene, (8) initiating mitigating activities, and (9) initiating recovery actions when directed. At the CVDF, the BED is a certified operations shift manager.

A listing of the primary and alternate BEDs by title, work location, and work telephone numbers is contained within the CVDF building emergency plan. The BED is on the CVDF premises during hazardous operations and is available through an "on-call" list 24 hours a day at all other times. Operations maintains a listing of on-call BED names, with work and home telephone numbers, at the Occurrence Notification Center.

The Incident Command Organization is also composed of a Hanford Fire Department Operations Section Chief (assigned by the Incident Commander), trained support staff (including Health Physics and Industrial Hygiene staff), and (as required) Hanford Fire Department medical responders and Hanford Patrol. In addition, accountability aides are responsible for facilitating the implementation of protective actions (evacuation or take cover) and for facilitating the
accountability of personnel after the protective actions have been implemented. Staging area managers are responsible for coordinating/conducting activities at the staging area. Personnel accountability aides assist the staging area managers by ensuring that personnel and visitors are properly evacuated from designated staging areas to a safe location. The Incident Command Organization supports actions requested by the Incident Commander and the BED.

B15 4 1 3 U.S. Department of Energy Hanford Emergency Operations Center As stated in Section 15 4 1 3 of the SNF Project FSAR, the DOE Hanford Emergency Operations Center is an emergency response facility, maintained by DOE, Richland Operations Office/Office of River Protection, to convene personnel providing essential emergency response functions. The DOE Hanford Emergency Operations Center will be activated for Alert and higher emergencies at the CVDF.

B15 4 2 Assessment Actions

Provisions of Section 15 4 2 of the SNF Project FSAR cover development and implementation of the hazards survey and hazards assessment, consequence assessments, and monitoring activities. These provisions apply to the CVDF.

The CVDF hazards survey will identify the conditions contained in the comprehensive emergency management program. A CVDF emergency planning hazards assessment will be developed for the CVDF for hazards that have the potential to generate an Alert or higher emergency. The hazards assessment will be prepared from the hazards survey and safety analyses that are developed and summarized in Chapter B3 0. The hazards assessment will also be derived from other pertinent facility documentation (e.g., safety assessment documents, interim safety basis documents, and special nuclear material accountability documents). The hazards assessment will provide the technical basis for the emergency management program. The scope and extent of planning and preparedness will directly correspond to the type and scope of hazards present and the potential consequences of events.

The hazards assessment will identify and characterize the hazards relevant to potential CVDF operational emergencies. This includes determination of the following:

- A broad range of initiating events
- Accident mechanisms
- Equipment or system failures
- Event indications
- Contributing events
- Source terms
- Material release characteristics
- Topography
- Environmental transport and diffusion
Annex B — Cold Vacuum Drying Facility

- Exposure considerations
- Chemical hazards

The hazards assessment will characterize the potential consequences to workers, the public, and the environment for each postulated accident and determine the emergency planning zone (EPZ) for each facility. The assessment will also determine the emergency class, protective actions, and observable indications and criteria (emergency action levels) corresponding to the range of identified accidents.

A spectrum of potential accidents ranging from minor to beyond design basis are postulated and will be realistically analyzed for the CVDF. While not every conceivable situation will be analyzed, the hazards assessments will provide the framework for response to virtually any declared emergency.

The methodology, assumptions, models, and evaluation techniques used in the hazards assessments are documented in Sections 15.4.2.1 and 15.4.2.2 of the SNF Project FSAR. Results from the CVDF hazards assessment will be used to develop the CVDF building emergency plan. Hazards assessments will be reviewed at least annually and updated, as necessary, in accordance with Section 15.4.2 of the SNF Project FSAR, to delineate significant changes to the CVDF or hazardous inventories, and be maintained in accordance with the site contractor document control requirements.

**B15.4.3 Event Categorization and Classification**

Event categorization and classification for CVDF emergency events will be accomplished in accordance with Section 15.4.3 of the SNF Project FSAR.

**B15.4.4 Notifications**

Notifications, in the event of an emergency event at the CVDF, will be made in accordance with the provisions of Section 15.4.4 of the SNF Project FSAR in order to mitigate consequences and to protect the health and safety of workers, the public, and the environment.

**B15.4.5 Emergency Facilities and Equipment**

The building emergency plan for the CVDF will be prepared and issued in accordance with HNF-IP-0263-GEN, *Building Emergency Plan Guidance*. A description of the facilities that will be available for coordinating CVDF emergency response activities will be specified in the building emergency plan.
Emergency equipment consisting of materials and tools that may be required to measure, control, or mitigate the consequences of an emergency at the CVDF is provided in Table B15-1. Detection ranges and types of instruments for radiological and nonradiological hazardous materials will be adequate for CVDF emergency conditions, as determined in Section B15 4.2. The emergency planning organization will ensure that sufficient emergency equipment is available. Location of this emergency equipment will be stated in the CVDF building emergency plan.

**B15 4.6 Protective Actions**

Protective actions are those actions that will be taken to preclude or reduce the exposure of individuals or the environment impacted by hazards or unsafe conditions during an emergency event at the CVDF. These protective actions are presented in Section 15 4.6 of the SNF Project FSAR and are applicable to the CVDF. Protective actions for the CVDF will reflect use of the emergency response planning guidelines identified in Section 15 4.3.1 of the SNF Project FSAR. The planning guidelines published in the Emergency Response Planning Guidelines (AIHA 1988) will be used during a CVDF emergency response to determine protective actions for unique exposures to chemical releases (see Table 15-5 of the SNF Project FSAR). The protective action guides are also used during an emergency response to determine protective actions for unique exposures to radiological releases (see Table 15-2 of the SNF Project FSAR). Protective action guides adopted by the states of Washington and Oregon (EPA-400) are published in DOE/RL-94-02, Hanford Emergency Management Plan.

The Hanford Site emergency management program uses the EPZ concept to focus emergency planning activities. EPZs are designated areas where protective actions may be required. The size of a zone is determined primarily by the expected dispersion distance of a particular concentration of a substance. The two exposure pathways for both radiological and nonradiological hazardous materials are the plume exposure pathway and the ingestion exposure pathway. See Section 15 4.6 of the SNF Project FSAR for a description of the exposure pathways. Figure B1-1 indicates the location of the K Basins and CVDF (also see Figure B1-12).

The plume exposure pathway EPZ is the probable area of exposure to a passing cloud, or plume, of the substance potentially resulting in direct contact with the substance through the exterior of the body or through inhalation. The plume exposure pathway EPZ includes the area where emergency planning is conducted (1) to ensure that prompt and effective actions are taken in the event of an emergency, (2) to protect onsite personnel, and (3) to ensure public health and safety. The plume exposure pathway for the CVDF (5 m) is shown in Table 15-6 and Figure 15-5 in the SNF Project FSAR.
Table B15-1  Cold Vacuum Drying Facility Emergency Equipment  (2 sheets)

<table>
<thead>
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<th>Type</th>
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<tbody>
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<td><strong>Fixed and portable equipment</strong></td>
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| Fire control system  
- Fire detection and alarm system  
and wet pipe automatic sprinkler suppression system | Assists in the control of a fire  
Assists in notifying personnel, summoning the Hanford Fire Department, and in fire suppression |
| Decontamination shower | Assists with personnel decontamination of hazardous (chemical) materials |
| Fire system pressure alarms and/or water flow alarm | Assists with notifying personnel of emergency conditions |
| Evacuation and take cover siren | Assists with notifying personnel of emergency conditions and, by the type of siren, expected actions |
| Red crash alarm telephone | Alerts personnel in an emergency and communicates emergency information |
| **Portable emergency equipment** | |
| Fire extinguisher  
(Types A, B, and C) | Assists in fire suppression |
| Hazardous materials spill control kits  
(unmounted) | Assists with hazardous (chemical) materials stabilization and cleanup following a spill or release |
| Command post equipment emergency procedures, checklists (maps and photographs of facilities optional) | Provides area and site-specific emergency information |
| Operational event scene equipment radiological response vehicle emergency procedures, duty cards, checklists, maps, photographs of facilities | Assists in controlling and mitigating the event |
Table B15-1  Cold Vacuum Drying Facility Emergency Equipment  (2 sheets)

<table>
<thead>
<tr>
<th>Type</th>
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<tr>
<td>Protective clothing and equipment</td>
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<tr>
<td>Anti-C clothing and personal protective equipment</td>
<td>Provides contamination control (anti-C clothing for radiological and acid gear for any corrosive chemicals)</td>
</tr>
<tr>
<td>Miscellaneous respiratory equipment</td>
<td>Provides respiratory protection for radionuclides, this type of respirator equipment is not considered to be emergency equipment</td>
</tr>
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</table>

The ingestion exposure pathway EPZ is the probable area of exposure to contaminated foodstuffs or water potentially resulting in deposition of the material in various internal organs following ingestion (eating or drinking). The ingestion exposure pathway EPZ for radiological and nonradiological incidents at all Hanford Site facilities corresponds to the 80-km (50-mi) EPZ for Energy Northwest’s (formerly Washington Public Power Supply System) Nuclear Plant 2. The circle in Figure 15-6 in the SNF Project FSAR represents the ingestion EPZ for the Hanford Site.

The protective actions required to minimize the exposure of workers and the public are summarized in Section 15.4.6 of the SNF Project FSAR. Examples of protective actions as a function of accident category and consequences are illustrated in Table 15-7 in the SNF Project FSAR.

B15 4.7 Training and Exercises

Emergency organizations for the CVDF will be formed, trained, and tested in accordance with the provisions of Section 15.4.7 of the SNF Project FSAR. Drills and exercises will be developed in accordance with the provisions of Section 15.4.7 of the SNF Project FSAR with sufficient scope and detail to emphasize the facility-specific emergency events and response actions applicable to the CVDF.

B15 4.8 Reentry and Recovery

The provisions applicable to a CVDF emergency event termination, facility entry, transition from an emergency organization to a recovery organization, and the recovery process are provided in Section 15.4.8 of the SNF Project FSAR.
B15 5 DOCUMENT CONTROL

The CVDF building emergency plan, implementing procedures, reports of drills and exercises, and emergency event documentation will be controlled and updated in accordance with the provisions of Section 15 5 of the SNF Project FSAR.

B15 6 REFERENCES


CHAPTER B16 0

PROVISIONS FOR DECONTAMINATION AND DECOMMISSIONING
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B16 3 DESCRIPTION OF CONCEPTUAL PLANS B16-1
   B16 3 1 Design Features B16-2
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LIST OF TERMS

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<td>Cold Vacuum Drying Facility</td>
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<tr>
<td>D&amp;D</td>
<td>decontamination and decommissioning</td>
</tr>
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<td>FSAR</td>
<td>final safety analysis report</td>
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<tr>
<td>HEPA</td>
<td>high-efficiency particulate air (filter)</td>
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<td>MCO</td>
<td>multi-canister overpack</td>
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<td>PWC</td>
<td>process water conditioning</td>
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PROVISIONS FOR DECONTAMINATION AND DECOMMISSIONING

INTRODUCTION

The provisions that apply to future decontamination and decommissioning (D&D) of Spent Nuclear Fuel (SNF) Project facilities are addressed in Chapter 16 0 of the SNF Project Final Safety Analysis Report (FSAR). Provisions specific to the Cold Vacuum Drying Facility (CVDF) are addressed in this Chapter B16 0.

The CVDF has been given a final designation of a hazard category 2 nuclear facility because of the potential for release of the radioactive inventory contained in the temporarily housed multi-cask overpacks (MCOs). The facility and a majority of the process systems will not contain significant, if any, contamination once the MCOs have been processed and delivered to the Canister Storage Building for interim storage. Facility hazards will be relatively low after MCO processing, and the CVDF processing equipment is not complex and is easy to access and decontaminate. Once the CVDF inventory has been removed, the facility process systems will be deactivated and the CVDF transitioned to the D&D program at the end of its useful life. The end state of the CVDF is not known at this time, but several options, ranging from facility reuse to complete dismantlement and removal, are available.

REQUIREMENTS

The requirements that form the basis for D&D provisions are identified in Section 16 2 of the SNF Project FSAR.

DESCRIPTION OF CONCEPTUAL PLANS

This section describes the design features and operational considerations pertinent to the CVDF that will facilitate decontamination and ultimate decommissioning and environmental restoration activities. Final decontamination activities are simplified when design and operations are combined to limit and identify the residual quantities and form of contamination, and the structures, systems, and components that have been exposed to or contaminated with material are identified. The design features that are of importance to D&D will be controlled through the design change control process to ensure that changes to the facility consider the impact on D&D. General considerations applicable to all SNF Project facilities are provided in Section 16 3 of the SNF Project FSAR.

The MCOs, the MCO transportation cask, and the CVDF are designed to operate so the facility is free from radioactive contamination and to also prevent the spread of radioactive contamination from inside the MCOs. Baseline operations for the CVDF ensure that the spread of radiological contamination from the MCOs or the transportation cask to the interior surfaces of...
the CVDF process areas is minimal during normal operations. The fuel loading operation is designed to minimize MCO and transportation cask contamination during fuel loading.

**B16 3 1 Design Features**

The CVDF has several features that are designed to control the radiological materials of the cold vacuum drying process and preclude the spread of these materials into nonprocess equipment and areas. These design features include confinement, compartmentalized systems and equipment, and selected safety features for prevention and mitigation of the postulated accidents identified in Chapter B3.0.

The ventilation systems are designed such that normally contaminated air is routed to only one system, the process bay local exhaust HVAC and process vent system, to minimize the expected amount of ventilation ducting for decontamination. During normal operation, this system receives contaminated air from the process bay hoods, process vents, MCO vents, and/or process water system tank vents. All other ventilation systems receive contaminated air only under abnormal operations and/or accident conditions. During an accident event, backflow preventers provided in the ducting prevent cross-contamination of other areas through the ventilation system.

During transportation to the CVDF, confinement for the SNF consists of (1) the MCO, and (2) the transportation cask. At the CVDF, the tractor-trailer is decoupled, leaving the trailer with the loaded cask–MCO in one of the four bays for processing. At this time, the cask lid is removed temporarily breaching secondary confinement, and a process hood is installed over the cask–MCO to provide contamination control.

Primary confinement acts as a barrier separating the material contained in the MCO and the process piping from surrounding areas. The CVDF primary confinement of liquid and airborne effluents consists of the MCO and the process piping, vessels and equipment in the vacuum purge system, and the process water conditioning (PWC) system connected to the MCO via the process hood. Secondary confinement is provided by the building structures in conjunction with the process general supply/exhaust HVAC system. Secondary confinement ventilation air is also exhausted through a dual-stage HEPA filter prior to release through the exhaust stack. For further description of the confinement philosophy, refer to Chapter B2.0.

**B16 3 1 1 Compartmentalized and Modular Systems and Equipment**

**B16 3 1 1 1 Multi-Canister Overpack** Contamination from routine operations or from accident conditions within the processing bay enclosures is precluded by providing a hood over the cask–MCO during opening and process connection activities and by designing processing...
systems to be a primary confinement boundary against the release of liquid or airborne contaminants from the MCO. The MCO's rupture disk relief and vent paths are vented through the process hood to the process bay local exhaust HVAC and process vent system HEPA filter to direct any overpressure or accident release through a filtered source instead of into the processing bay.

**B16 3 1 1 2 Process Bays** The design of the CVDF provides enclosed processing bay confinements during operation. The processing bays remain free from contamination during normal operation and would only be contaminated by an accidental release. Contamination of the processing bays is expected to be minimal during abnormal operations and during some accident conditions. The design of the process systems (the vacuum purge system, the PWC system, and the tempered water (annulus) system) precludes the spread of contamination from the cask-MCO to the bay during normal operations. The vacuum purge system and tempered water (annulus) system are contained on relatively compact skids that contain all necessary equipment and piping for system operation. To contain small leaks, the process equipment skid contains a retention basin covering the full bottom of the skid. In addition, each process bay contains a sump to collect any potentially contaminated liquid.

An epoxy coating covers the concrete floors and extends 4 ft up the walls of the bays to provide protection and ease of decontamination. The sumps will be protected with a stainless steel liner. Surveys and periodic decontamination of the bays may take place as necessary to maintain processing enclosures as clean as possible. The operating area floor is specified to have a sealer-hardener applied to its trowel-finished surface. This is sufficient to meet the intent of the ANSI/ANS-57.9-1992, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)*, requirement in paragraph 6 14(1) to provide concrete with an impervious surface.

**B16 3 1 1 3 Process Water Tank Room** The PWC system feeds the MCO process water to the process water tank room, which contains the receiver tanks, pumps, ion exchange modules, automatic sampling equipment, and a temporary storage tank that is used prior to transferring the process water by tanker truck to the K West Basin integrated water treatment system. All PWC system components, including connecting pipe, are located within the confinement of the process bays or the process water tank room. Unlike the process bays, the process water tank room may receive a minimal release from the receiver tanks during tank pressure relief conditions; the release is exhausted to the room through a HEPA filter. All releases will be contained within the process water tank room.

Piping is installed to drain and not normally retain fluid. To contain small leaks, the PWC skid contains a catch pan 1-in deep covering the full bottom of the skid. The process water tank room is designed as a sump to contain any liquid releases that may occur. Access to the room is only through the transfer corridor and change room to prevent the spread of contamination throughout the facility. The concrete floors and walls are designed to retain liquids and prevent leakage to below grade from any liquid spills. An epoxy coating covers the concrete floors and extends 4 ft feet up the walls of the process water tank room for protection and ease of decontamination, in case contamination occurs.
B16 3 1 2 Safety Features  Chapters B3 0, B4 0, B5 0, and B6 0 identify the selected safety features for preventing or mitigating the postulated accidents (see tables identified in Chapter B3 0 for each accident) These implemented engineered barriers and administrative programs will prevent or greatly reduce the consequences of any postulated upset event or accident, thus the amount and location of any radiological material releases inside the CVDF will be diminished for D&D cleanup activities  No design basis accident will result in offsite dose consequences (prevented) or onsite dose consequences exceeding the evaluation guidelines (mitigated)

B16 3 2 Operational Considerations

The potential for personnel, equipment, and building contamination within the CVDF is minimized by the design of the facility and equipment, and by administrative controls, radiological practices, and work guidelines defined in operating procedures and work permits  The general design features and operating practices described in Sections 16 3 1 and 16 3 2 of the SNF Project FSAR were considered in the design of the CVDF to minimize the spread of contamination, simplify D&D operations, and help minimize site and environmental contamination

Because baseline operations assume no spread of contamination from the cask-MCO or process equipment, no special facilities for the support of decontamination activities have been provided  Although the risk of contamination is minimal, operating procedures address requirements for radiological control surveys in various areas of the facility and at specified points in the operating activities  For example, surveys will be performed to verify that the cask-MCO is received from K Basins with the exterior free of contaminations  In addition, the process bay enclosure and the cask transport system are surveyed and decontaminated as necessary before opening the bay to remove a cask for transport  This will minimize the spread of contamination outside the bay enclosure during entry and exit of the transport vehicle  Surveys and periodic decontamination of the process water tank room may take place as necessary to maintain processing enclosures as clean as possible

The secondary confinement will be operated so that it is held at a negative pressure under normal operations and accident conditions  Negative pressure will ensure that air leakage flows from the noncontaminated outside environment to the potentially slightly contaminated systems and areas

Additional operational considerations to facilitate D&D include the following

- The process equipment will be leak tested prior to each use
- System flushes will take place in the drain systems after each draining step
Process bay sumps will be sampled whenever liquid is detected, and the sump content will be appropriately disposed based on the results of the sample.

During CVDF operations, site environmental contamination is minimized because of the following protection features:

- No expected liquid releases
- HEPA-filtered final exhaust
- Negative building pressure maintained by the heating, ventilation and air conditioning system

**B16 3 3 Decommissioning**

The CVDF and its process equipment may be removed at a future time in compliance with applicable regulations. See Section 16.3.3 of the SNF Project FSAR for the process used to develop the CVDF D&D plan. Conceptual plans for D&D will include an updated facility hazard analysis for the D&D activities, which will be used to prepare the plan to administer the expected D&D strategy. Conceptual plans also will include a preliminary deactivation plan that contains at least information about:

- Structures, systems, and components in their final configuration
- Review and determination of status for structures, systems, and components based on mission and life cycle phase
- Configuration management for missing or inaccurate design baseline documentation, voiding and downgrading of design documents, and turnover of design baseline documents to the environmental restoration contractor

Decommissioning plans for the CVDF facility will be developed and reviewed against the existing environmental impact statement. The environmental impact statement will be updated to include D&D activities in accordance with the *National Environmental Policy Act* process, if deemed appropriate.

The construction of the CVDF processing areas as an above-grade steel frame and prestressed concrete panel structures, and the process support and administrative areas as metal frame and siding buildings, will simplify decontamination and dismantling. The dose resulting from decommissioning has been limited in a number of ways. One is by reducing the amount of residual radioactive material within process and ventilation equipment. This is accomplished primarily by the use of surfaces made of smooth, nonporous materials with a minimum of cracks or crevices. For example, the process water receiver tanks have conical bottoms in addition to...
being made with an electro-polished interior surface, and ventilation ducts are designed with runs as short as possible. In addition, the receiver tanks and the PWC filters have flushing provisions.

A second method of reducing the decommissioning dose is by the use of skid-mounted processing equipment connected to other equipment by flanged couplings rather than welding. Also, process piping is located to provide easy access. These features limit the amount of time required to perform dismantling and the amount of time that workers will be required to work in areas where exposure will be received. In addition, components with the highest potential for containing radioactivity are segregated in the process water tank room, and the PWC ion exchange modules and filters are provided with integral shielding. Removing such components with their shielding as one unit will reduce the time workers will need to be in a radiation field, while the shielding will continue to protect them during the activity.

To reduce contamination, the floors and bottom 4 ft of the facility walls are coated with epoxy coatings.

**B16 4 REFERENCES**


*National Environmental Policy Act (NEPA) of 1969, 42 USC, 4321, et seq*
CHAPTER B17 0

MANAGEMENT, ORGANIZATION, AND INSTITUTIONAL SAFETY PROVISIONS
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B17.1 INTRODUCTION
B17.2 REQUIREMENTS
B17.3 ORGANIZATIONAL STRUCTURE, RESPONSIBILITIES, AND INTERFACES
B17.4 SAFETY MANAGEMENT POLICIES AND PROGRAMS
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<th>Description</th>
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B17 0 MANAGEMENT, ORGANIZATION, AND INSTITUTIONAL SAFETY PROVISIONS

B17 1 INTRODUCTION

A description of the organizational structure, responsibilities, and interfaces that support safe design, construction, and operational activities of the Cold Vacuum Drying Facility (CVDF) as a subproject of the Spent Nuclear Fuel (SNF) Project are addressed in Chapter 17 0 of the SNF Project Final Safety Analysis Report (FSAR).

B17 2 REQUIREMENTS

The requirements that form the basis for management, organizational, and safety provisions are identified in Section 17 2 of the SNF Project FSAR.

B17 3 ORGANIZATIONAL STRUCTURE, RESPONSIBILITIES, AND INTERFACES

The overall organizational structure, responsibilities, and interfaces for CVDF operations are identified in Section 17 3 of the SNF Project FSAR.

B17 4 SAFETY MANAGEMENT POLICIES AND PROGRAMS

The safety management policies and programs applicable to CVDF are identified in Section 17 4 of the SNF Project FSAR.
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Annex D-200 Area Interim Storage Area Final Safety Analysis Report

Prepared for the U.S. Department of Energy
Assistant Secretary for Environmental Management

Project Hanford Management Contractor for the
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Richland, Washington

EDT: 626891

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VOLUME 5

ANNEX D – 200 AREA INTERIM STORAGE AREA

FINAL SAFETY ANALYSIS REPORT

January 2000
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EXECUTIVE SUMMARY

DE.1 FACILITY BACKGROUND AND MISSION

The 200 Area Interim Storage Area (200 Area ISA) at the Hanford Site provides for the interim storage of non-defense reactor spent nuclear fuel (SNF) housed in aboveground dry cask storage systems. The 200 Area ISA is a relatively simple facility consisting of a boundary fence with gates, perimeter lighting, and concrete and gravel pads on which to place the dry storage casks. The fence supports safeguards and security and establishes a radiation protection buffer zone. The 200 Area ISA is nominally 200,000 sq. ft. and is located west of the Canister Storage Building (CSB). Interim storage at the 200 Area ISA is intended for a period of up to 40 years until the materials are shipped off-site to a disposal facility. This Final Safety Analysis Report (FSAR) does not address removal from storage or shipment from the 200 Area ISA.

Three different SNF types contained in three different dry cask storage systems are to be stored at the 200 Area ISA, as follows:

- **Fast Flux Test Facility Fuel** – Fifty-three interim storage casks (ISC), each holding a core component container (CCC), will be used to store the Fast Flux Test Facility (FFTF) SNF currently in the 400 Area.

- **Neutron Radiography Facility (NRF) Training, Research and Isotope Production, General Atomics (TRIGA¹) Fuel** – One Rad-Vault² container will store

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¹ TRIGA is a registered trademark of Gulf General Atomics Company, Inc.

² Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.
two U.S. Department of Transportation (DOT)-6M containers and six NRF TRIGA casks currently stored in the 400 Area.

- **Commercial Light Water Reactor Fuel** – Six International Standards Organization (ISO) containers, each holding a Nuclear Assurance Corporation (NAC)-1 cask with an inner commercial light water reactor (LWR) canister, will be used for commercial LWR SNF from the 300 Area.

An aboveground dry cask storage location is necessary for the spent fuel because the current storage facilities are being shut down and deactivated. The spent fuel is being transferred to interim storage because there is no permanent repository storage currently available.

**DE.2 FACILITY OVERVIEW**

The 200 Area ISA is located within the Hanford Site 200 East Area and will provide safe outside storage of the SNF, while protecting the fuel through the use of storage systems resistant to man-made and natural phenomena hazards. The majority of the fuel to be stored within the 200 Area ISA will consist of FFTF SNF. However, the 200 Area ISA will also store other SNF from the Hanford Site, including NRF TRIGA fuel from the 400 Area and commercial LWR fuel, currently stored in the 300 Area.

The 200 Area ISA is located a few hundred feet west of the CSB. The footprint of the ISA is nominally 500 ft. by 400 ft., surrounded by a 7-ft. chain-link fence topped with barbed wire. Five gates in the fence control access of vehicles and personnel. Lighting is provided on the perimeter. Within the fenced area are two concrete pads for placement of ISCs and one concrete pad for the ISO containers. The Rad-Vault container will be placed on level gravel.

An equipment storage building (pre-engineered metal building) is proposed near the 200 Area ISA to house lifting devices, impact limiters, transfer casks, and various other types of
equipment associated with storage, movement, or transport of casks. This storage building will be located outside the 200 Area ISA fence and provides no safety-related function.

The 200 Area ISA facility components (excluding the dry cask storage systems) are classified as General Service; the dry cask storage systems are designated Safety Significant. The CCCs and the LWR canisters are designated Safety Class for criticality geometry control.

No uncontained radioactive materials will be handled at the storage area. Therefore, decontamination and decommissioning efforts should be minimal.

There is interaction of the ISA with existing CSB facilities for surveillance activities.

DE.3 FACILITY HAZARD CLASSIFICATION

A final hazards categorization of the 200 Area ISA facility was performed based on the final hazard analysis (SNF-4820) and accident analyses documentation (Chapter D3.0) for the facility. Consistent with DOE-STD-1027-92, Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports, the final categorization was based on the material-at-risk quantities identified in an individual cask inventory. The ISA material-at-risk quantities were compared against the DOE-STD-1027-92 threshold quantities. The ISA facility final hazard categorization found the ISA facility to be a Hazard Category 2 facility. This categorization level is consistent with the bases and guidance described in DOE-STD-1027-92. Hypothetical release source terms were developed for each fuel type. The 200 Area ISA inventory assumed for the analysis is based on a per cask basis and either includes seven FFTF driver assemblies, all 101 TRIGA elements, or one LWR assembly.

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DE.4 SAFETY ANALYSIS OVERVIEW

The 200 Area ISA is a passive storage facility designed to hold up to 70 storage containers of SNF for up to 40 years. Potential hazards associated with receipt handling and storage of these containers are the release of radioactivity, loss of shielding, or loss of configuration control for criticality. Seven design basis accidents (DBAs) were identified and analyzed to evaluate the potential consequences of these hazards to on-site workers and the public. The seven DBAs analyzed are as follows:

- Cask handling/drop
- Mobile crane mechanical failure
- Cask tip over
- Fuel rod rupture
- Seismic event
- Tornado/wind
- Fire.

In all but one case, the consequences for all three fuel types were prevented by the passive design features of the storage systems. In the case of the mobile crane mechanical failure, credit was not taken for the integrity of the NRF TRIGA storage systems and an unmitigated release was assumed. The consequences for this accident and fuel type were found to be significantly below both the off-site limits and the on-site guidelines.

Preventive and mitigative aspects relied on in the accident analyses are shown in Table DES-1.
# Table DES-1. Structure, System, and Component Safety Classification Summary.

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(1) This item is designated as Safety Class for structural integrity to provide criticality geometry control.
(2) This item upgraded to Safety Significant to accomplish NRC equivalency based on important-to-safety Category B classification.
(3) Critical lift requirements imposed using DOE/R1-92-36, *Hanford Site Hoisting and Rigging Manual*, to accomplish NRC equivalency for important-to-safety Category B.

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CCC = core component container.  
DOT = U.S. Department of Transportation.  
FFTF = Fast Flux Test Facility.  
GS = general service.  
ISC = interim storage cask.  
ISO = International Standards Organization.  
LWR = light water reactor.  
NA = not applicable.  
NAC = Nuclear Assurance Corporation.  
NRC = U.S. Nuclear Regulatory Commission.  
NRF = Neutron Radiography Facility.  
SAR = safety analysis report.  
SC = safety class.  
SS = safety significant.  
TRIGA = Training, Research and Isotope Production, General Atomics.
DE.5 ORGANIZATIONS

Fluor Hanford is responsible to the DOE for planning, integrating, and managing SNF Project activities, including programs, projects, and operations. Organizational responsibilities related to the SNF Project are summarized in the executive summary, and described in detail in Chapter 17.0, of the SNF Project FSAR.

DE.6 SAFETY ANALYSIS CONCLUSIONS

This FSAR Annex concludes that the 200 Area ISA can be operated safely without exceeding the off-site limits or on-site guidelines.

One remaining open issue is that the NAC-1 inner LWR canister design is not yet approved. The analyses in this SAR are based on a canister design that is in the final stages of design, review, and approval. The design provides a welded closure and dimensional constraints to accommodate future packaging in standard containers. Characteristics of the canister (e.g., design pressures or structural strength) are expected to be unchanged from the canister design used as the basis for evaluation. Calculations and text in this section will be verified upon design completion and approval of a design analysis report for the LWR canister. Revision will be required only if the characteristics and features of the canister are changed significantly. All affected discussions will be reviewed for applicability based on the final LWR canister design.

DE.7 FINAL SAFETY ANALYSIS REPORT ORGANIZATION

The 200 Area ISA FSAR is based on the format and content guidance of DOE-STD-3009-94, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports, and the requirements of DOE Order 5480.23, Nuclear Safety

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Analysis Reports. This report also includes content guidance from NRC Regulatory Guide 3.26, Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants, as a result of the DOE regulatory policy described in HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements. Due to the similar mission of the ISA and the CSB for SNF dry storage, HNF-SD-SNF-SP-012, Additional Guidance for Including Nuclear Safety Equivalency in the Canister Storage Building and Cold Vacuum Drying Facility Final Safety Analysis Reports, supplemental requirements for the CSB (Table 3) were used for guidance in the preparation of SNF-3446, Spent Nuclear Fuel Project - Criteria Document, Spent Nuclear Fuel Final Safety Analysis Report, Rev. 1, Appendix E, “FSAR Format and Content - Volume 5: Annex D, 200 Area Interim Storage Area FSAR.” Annex D was written in accordance with the criteria document (SNF-3446).

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<tr>
<td>AC</td>
<td>Administrative Control</td>
</tr>
<tr>
<td>ACI</td>
<td>American Concrete Institute</td>
</tr>
<tr>
<td>AFFRI</td>
<td>Armed Forces Fuel Research Institute</td>
</tr>
<tr>
<td>AL</td>
<td>aluminum</td>
</tr>
<tr>
<td>ALARA</td>
<td>as low as reasonably achievable</td>
</tr>
<tr>
<td>ANS</td>
<td>American Nuclear Society</td>
</tr>
<tr>
<td>ANSI</td>
<td>American National Standards Institute</td>
</tr>
<tr>
<td>ASCE</td>
<td>American Society of Civil Engineers</td>
</tr>
<tr>
<td>ASME</td>
<td>American Society of Mechanical Engineers</td>
</tr>
<tr>
<td>ASTM</td>
<td>American Society for Testing and Materials</td>
</tr>
<tr>
<td>ATM</td>
<td>approved testing material</td>
</tr>
<tr>
<td>AWG</td>
<td>American wire gauge</td>
</tr>
<tr>
<td>BDBA</td>
<td>beyond design basis accident</td>
</tr>
<tr>
<td>BED</td>
<td>building emergency director</td>
</tr>
<tr>
<td>BR</td>
<td>breathing rate</td>
</tr>
<tr>
<td>BWR</td>
<td>boiling water reactor</td>
</tr>
<tr>
<td>CCC</td>
<td>core component container</td>
</tr>
<tr>
<td>CCTV</td>
<td>closed circuit television</td>
</tr>
<tr>
<td>CEDE</td>
<td>committed effective dose equivalent</td>
</tr>
<tr>
<td>CFR</td>
<td>Code of Federal Regulations</td>
</tr>
<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
</tr>
<tr>
<td>CSER</td>
<td>criticality safety evaluation report</td>
</tr>
<tr>
<td>D&amp;D</td>
<td>decontamination and decommissioning</td>
</tr>
<tr>
<td>DBA</td>
<td>design basis accident</td>
</tr>
<tr>
<td>DBE</td>
<td>design basis earthquake</td>
</tr>
<tr>
<td>DBF</td>
<td>design basis fire</td>
</tr>
<tr>
<td>DBT</td>
<td>design basis tornado</td>
</tr>
<tr>
<td>DFA</td>
<td>driver fuel assemblies</td>
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<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
</tr>
<tr>
<td>DOE-RL</td>
<td>U.S. Department of Energy, Richland Operations Office</td>
</tr>
<tr>
<td>DORF</td>
<td>Diamond Ordnance Reactor Facility</td>
</tr>
<tr>
<td>DOT</td>
<td>U.S. Department of Transportation</td>
</tr>
<tr>
<td>DP</td>
<td>daughter product</td>
</tr>
<tr>
<td>EOC</td>
<td>Emergency Operations Center</td>
</tr>
<tr>
<td>EPZ</td>
<td>emergency planning zone</td>
</tr>
<tr>
<td>ERPG-2</td>
<td>Emergency Response Planning Guide-2</td>
</tr>
<tr>
<td>FFCR</td>
<td>fuel follower control rod</td>
</tr>
<tr>
<td>FFTF</td>
<td>Fast Flux Test Facility</td>
</tr>
<tr>
<td>FHA</td>
<td>fire hazards analysis</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>GRC</td>
<td>galvanized rigid conduit</td>
</tr>
</tbody>
</table>
LIST OF TERMS cont.

GS  general service
HSRCM  Hanford Site Radiological Controls Manual
HWVP  Hanford Waste Vitrification Plant
ID  inner dimension
JEM  interim examination and maintenance
ISA  interim storage area
ISC  interim storage cask
ISO  International Standards Organization
ITS  important to safety
LCO  Limiting Condition for Operation
LWR  light water reactor
MCC  Material Characterization Center
MCO  Multi-Canister Overpack
MLRS  Multiple Launch Rocket System
MOX  mixed oxide
MPFL  maximum possible fire loss
MTHM  metric ton of heavy metal
MTU  metric ton uranium
MWd  megawatt - day
NAC  Nuclear Assurance Corporation
NEMA  National Electrical Manufacturers Association
NFPA  National Fire Protection Association
NFS  Nuclear Fuel Services
NPH  natural phenomena hazard
NRC  U.S. Nuclear Regulatory Commission
NRF  Neutron Radiography Facility
NRMCA  National Ready Mix Concrete Association
NSNF  National Spent Nuclear Fuel
NTS  Nevada Test Site
OCRWM  Office of Civilian Radioactive Waste Management
OD  outer dimension
PFP  Plutonium Finishing Plant
PNNL  Pacific Northwest National Laboratory
PUREX  Plutonium-Uranium Extraction (Facility)
PVC  polyvinyl chloride
PWR  pressurized water reactor
QARD  quality assurance requirements and description
RCT  radiological control technician
REDOX  Reduction-Oxidation (Facility)
SAR  safety analysis report
SARP  safety analysis report for packaging
SC  safety class
<table>
<thead>
<tr>
<th>Acronym</th>
<th>Description</th>
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</thead>
<tbody>
<tr>
<td>SCBA</td>
<td>self-contained breathing apparatus</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
</tr>
<tr>
<td>SRP</td>
<td>segmented rod program</td>
</tr>
<tr>
<td>SS</td>
<td>stainless steel</td>
</tr>
<tr>
<td>SS</td>
<td>safety significant</td>
</tr>
<tr>
<td>SSC</td>
<td>structure, system, and component</td>
</tr>
<tr>
<td>TRIGA</td>
<td>Training, Research and Isotope Production, General Atomics</td>
</tr>
<tr>
<td>TSR</td>
<td>Technical Safety Requirement</td>
</tr>
<tr>
<td>TWRS</td>
<td>Tank Waste Remediation System</td>
</tr>
<tr>
<td>UBC</td>
<td>Uniform Building Code</td>
</tr>
<tr>
<td>UD</td>
<td>unit dose</td>
</tr>
<tr>
<td>WESF</td>
<td>Waste Encapsulation and Storage Facility</td>
</tr>
<tr>
<td>WNP</td>
<td>Washington Nuclear Plant</td>
</tr>
<tr>
<td>YMP</td>
<td>Yucca Mountain Project</td>
</tr>
<tr>
<td>ZPA</td>
<td>zero-period acceleration</td>
</tr>
</tbody>
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CHAPTER D1.0

SITE CHARACTERISTICS
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D1.0 SITE CHARACTERISTICS

D1.1 INTRODUCTION

The 200 Area Interim Storage Area (ISA) is in the 200 East Area of the U.S. Department of Energy (DOE) Hanford Site immediately to the west of the Canister Storage Building (CSB), as shown in Figure D1-1. The ISA will be used to provide interim storage of non-defense reactor spent nuclear fuel (SNF), as described in DOE/EA-1185, Environmental Assessment, Management, of Hanford Site Non-Defense Production Reactor Fuel, Hanford Site, Richland, Washington.

The objective of this chapter is to describe the characteristics of the site on which the ISA is located. This chapter and Chapter 1.0 of the SNF Project Final Safety Analysis Report (FSAR) contain information about regional and Hanford Site characteristics. A detailed description of the ISA structure and storage system is provided in Chapter D2.0. Chapter D1.0 supports the hazard analysis and accident analyses in Chapter D3.0.

D1.2 REQUIREMENTS

The requirements that establish the basis for ISA siting are identified in this chapter and in Section 1.2 of the SNF Project FSAR. A discussion of U.S. Nuclear Regulatory Commission (NRC) equivalency requirements is also provided in Section 1.2 of the SNF Project FSAR.

In addition to the pertinent requirements identified in Section 1.2 of the SNF Project FSAR, the following industry standards are applicable to the ISA safety basis:

- ASCE 7-93, Minimum Design Loads for Buildings and Other Structures

D1.3 SITE DESCRIPTION

A description of the Hanford Site and associated areas is provided in Section 1.3 of the SNF Project FSAR. The following sections address the geography, demography, and regional land and water use of the area encompassed by and surrounding the CSB and the adjacent ISA. Siting evaluations performed for the CSB are applicable to the adjacent ISA.
D1.3.1 Geography

D1.3.1.1 Hanford Site Vegetation. Information applicable to all SNF Project facilities is provided in Section 1.3.1.1 of the SNF Project FSAR.

D1.3.1.2 Hanford Site Facilities. Information applicable to all SNF Project facilities is provided in Section 1.3.1.2 of the SNF Project FSAR.

D1.3.1.3 Boundaries for Evaluation of Accident and Effluent Release Limits. The ISA is within the DOE-controlled zone. Consequences of accident releases from the ISA to collocated workers are calculated in Section D3.4.2 at 100 m from the point of release, in accordance with approved procedures. Routine and accidental releases to the public (off-site receptor) are calculated at the Site boundary shown in Figure D1-2. This is also the location of the controlled area boundary, as the term is defined in Title 10, Code of Federal Regulations, Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste,” Section 106, “Controlled Area of an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage” (10 CFR 72.106).

Radiological consequences calculated in Section D3.4.2 were developed considering only the current Hanford Site boundary. Consequences for a receptor located on Highway 240 were calculated for information purposes only (Scott 1995). DOE and its contractors can control access during emergency and accident conditions. This access control meets the requirements of 10 CFR 72.106(b). Before changes to the Site boundaries are placed into effect by the proper authorities, the calculations in this FSAR Annex will be reviewed (and reanalyzed if required) in accordance with DOE Order 5480.21, Unreviewed Safety Questions.

D1.3.2 Demography

A description of the demography surrounding the Hanford Site is provided in Section 1.3.2 of the SNF Project FSAR. Figure D1-3 shows the 1996 200 East Area on-site employee population.

D1.4 ENVIRONMENTAL DESCRIPTION

D1.4.1 Meteorology

Meteorology information applicable to all SNF Project facilities in provided in Section 1.4.1 of the SNF Project FSAR.

The specific air transport factors used for accident consequence analyses, and the basis for their calculation and use, are discussed in further detail in HNF-SD-SNF-TI-059, A
Discussion on the Methodology for Calculating Radiological and Toxicological Consequences for the Spent Nuclear Fuel Project at the Hanford Site, and in Section D3.4.1.2. Data is provided for ground-level releases. To support the accident analyses of Section D3.4.2, air transport factors (X/Q and X/Q'), representing the dilution of a contaminant by atmospheric turbulence and diffusion as the contaminant travels downwind, were calculated (HNF-SD-SNF-TI-059). The symbol X/Q is the ratio of the average air concentration at the receptor to the average release rate at the release point. This ratio is used to assess potential radiological dose and noncorrosive chemical concentration at downwind locations. The symbol X/Q is the normalized peak air concentration at the center of a puff, divided by the quantity released, and is used to assess the consequences to a receptor for corrosive chemicals. The X/Q's for the analyses were calculated using joint frequency distribution data so as to be exceeded only 0.5% of the time (99.5 percentile) for each sector, or to be exceeded only 5% of the time (95 percentile) for data from all sectors combined (the greater of the two calculated values is used in the analyses).

The GXQ computer code, Version 4.0 (WHC-SD-GN-SWD-30002), was used to generate X/Q and X/Q' values. GXQ incorporates the methods described in NRC Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.

The wind rose data (Figures D1-4 and D1-5) indicate that winds from the west-northwest sector occur most frequently (nearly 20% of the time). That is, the emissions are transported toward the east-southeast sector. Winds out of the northwest and west also occur with a relatively high frequency (12% and 11%, respectively). The information provided in these figures was obtained from data presented in HNF-SD-SNF-TI-059.

D1.4.2 Hydrology

Hanford Site hydrology information that is applicable to all SNF Project facilities is provided in Section 1.4.2 of the SNF Project FSAR.

D1.4.2.1 Surface Water. Surface water information is provided in Section 1.4.2.1 of the SNF Project FSAR.

D1.4.2.2 Vadose Zone. A definition of the vadose zone is provided in Section 1.4.2.2 of the SNF Project FSAR.

D1.4.2.3 Aquifers. The 200 Area aquifer systems are discussed in Section 1.4.2.3 of the SNF Project FSAR.
**D1.4.3 Geology**

A description of the Hanford Site geology that applies to all SNF Project facilities is provided in Section 1.4.3 of the SNF Project FSAR.

**D1.4.3.1 Physiographic Setting of the Hanford Site.** Information on the physiographic characteristics applicable to the ISA is provided Section 1.4.3.1 of the SNF Project FSAR.

**D1.4.3.2 Stratigraphy.** Information on the stratigraphy of the Pacific Northwest and the Hanford Site that is applicable to the ISA is provided in Section 1.4.3.2 of the SNF Project FSAR.

**D1.4.3.3 History of Cataclysmic Flooding in the Pasco Basin.** Information on geologic structures that is applicable to the ISA is provided in Section 1.4.3.3 of the SNF Project FSAR.

**D1.4.3.4 Geologic Structures of the Columbia Basin and Hanford Site.** Information on geologic structures that is applicable to the ISA is provided in Section 1.4.3.4 of the SNF Project FSAR.

**D1.4.3.5 Geology of the Interim Storage Area.** The geology of the 200 East Area is summarized in Figures D1-6 through D1-8. The suprabasalt sediments consist of the Ringold formation and Hanford formation. The Ringold formation conforms to the basalt bedrock surface and tilts southeast toward the axis of the Cold Creek syncline. The Ringold formation is dominated by gravel units E and A, which are separated by the Lower Mud unit. These are the main unconfined aquifers. The Ringold formation thins from 164 ft at the south end to nearly pinching out at the north end. The beds have been truncated and are unconformably overlain by the Hanford formation. The Hanford formation is mainly sands and gravelly sands that are between 246 and 328 ft thick at the site. Additional information on the geology at the ISA site is provided in Section 1.4.3 of the SNF Project FSAR.

**D1.4.3.6 Tectonic Development of the Hanford Site.** Information on geologic structures and faults that relate to the Hanford Site seismic hazard analysis that is applicable to the ISA is provided in Section 1.4.3.6 of the SNF Project FSAR. Additional information is provided in WHC-SD-W236-TI-002, *Probabilistic Seismic Hazards Analysis, DOE Hanford Site, Washington.*

**D1.4.3.7 Contemporary Stress and Strain.** Information on earthquake activity, contemporary stress measurements, and subsidence history that is applicable to the ISA (WHC-SD-W236-TI-002) is provided in Section 1.4.3.7 of the SNF Project FSAR.
D1.4.3.8 Geologic Hazards.

D1.4.3.8.1 Seismic Hazard Assessment. The ISA is a Hazard Category 2 facility, as stated in HNF-1755, Initial Hazard Classification for the 200 East Area Interim Storage Area Project W-518. Performance Category 3 is applicable for a Hazard Category 2 facility. Per WHC-SD-SNF-DB-009, Appendix C, 200 Area Interim Storage Area Natural Phenomena Hazards, the seismic loads for the ISA Performance Category 3 structures, systems, and components (SSCs) follow the DOE design requirements specified in HNF-PRO-097, Engineering Design and Evaluation, which implements DOE Order 6430.1A, General Design Criteria, Section 1300-3, “Safety Class Criteria.” This approach is consistent with the application of seismic criteria within the SNF Project. The Performance Category 3 horizontal and vertical design response spectra values were taken from Table 6, “Design Response Spectra for Performance Category 3 in the 100 and 200 Areas” of HNF-PRO-097. The values for various damping levels umbrella the 100 Area, 200 West Area, and 200 East Area and are slightly higher than 200 East Area-specific values (0.26 g rather than 0.24 g).

D1.4.3.8.2 Volcanic Hazard Assessment. Information about volcanic hazards applicable to all SNF Project facilities is provided in Section 1.4.3.8.2 of the SNF Project FSAR.

D1.4.3.8.3 Subsurface Stability. The ISA is constructed in flood sediments, the youngest sediments being approximately 13,000 years old. There are no areas of potential surface or subsurface subsidence, uplift, or collapse except for the low geologic deformation discussed in Section 1.4.3.6 of the SNF Project FSAR. With the exception of the loose, surficial, wind-deposited silt, soils are competent and form good foundations. Several geotechnical studies have been completed in and around existing tank farms. Liquefaction of soils beneath the tank farms is not a credible hazard because the water table is greater than 215 ft below ground surface. Liquefaction cannot occur in dry sediments. Liquefaction is also not a concern because of the results of evaluations discussed below for the Hanford Waste Vitrification Plant (HWVP), which is in close proximity to the ISA.

A geotechnical investigation was performed at the site in 1989 (Dames and Moore 1989) to evaluate subsurface conditions for the design and construction of the HWVP. The HWVP Project was canceled and the site is being used for the CSB. Subsurface conditions were investigated by 17 borings ranging in depth from 20 to 100 ft at the site and in the surrounding area. Geophysical tests, including a series of downhole seismic tests, were conducted in boring VP-8, approximately 400 ft from the ISA/CSB. Borehole VP-15 (Figure D1-9) is at the ISA/CSB site.

The 17 geotechnical investigation borings were advanced without drilling fluid by driving a 6-in. diameter, steel “core barrel” with a “split jar” downhole hammer. As the borings were advanced, an 8-in. diameter steel casing was driven to prevent caving of soils above the sampling depth. Water was occasionally poured into the hole to prevent caving of the loose, dry soils and to aid in recovering samples. Slightly disturbed soil samples were taken at 5-ft intervals by driving a sampler using either a 705-lb hammer falling through a distance of 18 in. or a 825-lb
hammer falling through a distance of 28 in. During the sampling process, resistance to penetration of the sampler, in blows per foot, was recorded.

The soil profile in the upper 100 ft consists essentially of the three strata shown in Figure D1-10. Borehole VP-14 is approximately 230 ft west of the ISA/CSB. The stratigraphy in VP-15 (Figure D1-9) is the same as at VP-14. The dynamic soil properties of the site are summarized in Figure D1-11. More detailed descriptions of the geotechnical investigations and interpretations are found in Report of Geotechnical Investigations for the Proposed Hanford Waste Vitrification Plant, Hanford Washington (Dames and Moore 1989) and in Section 1.4.3.8.2 of the SNF Project FSAR.

D1.5 NATURAL PHENOMENA THREATS

Section 2.3 of HNF-2524, 200 East Interim Storage Area Preliminary Safety Evaluation, states that recommendations for the classification of safety-class and safety-significant SSCs are derived from DOE Order 6430.1A and HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements. Design requirements for natural phenomena hazards are established in Engineering Change Notice 643545 to WHC-SD-SNF-DB-009, Canister Storage Building Natural Phenomena Hazards, and Section 2.3.2 of HNF-2524. A listing of safety-class and safety-significant SSCs required for the ISA are presented in Table D4-2. Table D1-1 presents a summary of natural phenomena design loads and criteria for ISA safety-class SSCs.

D1.6 EXTERNAL HUMAN-GENERATED THREATS

This section identifies and investigates specific potential human-generated threats to ISA operation. Threats to the ISA from human activities that are not known at this time will be evaluated when identified by the unreviewed safety question process.

D1.6.1 Aircraft Activity

The methodology used in aircraft activity analysis is provided in Section 1.6.1 of the SNF Project FSAR. There are nine active airports within a 24-mi radius of the ISA (HNF-1786). Eight of these are small airports that serve only general aviation aircraft. The Richland Airport, 21 mi southeast of the ISA, supports primarily general aviation operations, but two commercial freight carriers per day land and take off from the airport's runways. The nearest airport with significant commercial and military air activity is the Tri-Cities Airport, 29 mi southeast of the ISA.
Table D1-1. Interim Storage Area Safety-Class Natural Phenomena Design Loads.

<table>
<thead>
<tr>
<th>Hazard</th>
<th>Load</th>
<th>Design guidance</th>
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<tbody>
<tr>
<td>Seismic</td>
<td>Median response spectra: (1)</td>
<td>DOE Order 5480.28 (2)</td>
</tr>
<tr>
<td></td>
<td>0.26 g horizontal</td>
<td>DOE-STD-1020-94 (3)</td>
</tr>
<tr>
<td></td>
<td>0.17 g vertical</td>
<td>HNF-SD-SNF-DB-009, App C</td>
</tr>
<tr>
<td>Straight wind</td>
<td>80 mi/h, fastest mile at 30 ft</td>
<td>ASCE 7-93 (4)</td>
</tr>
<tr>
<td></td>
<td>Wind speeds</td>
<td>DOE-STD-1020-94 (3) (including missiles)</td>
</tr>
<tr>
<td></td>
<td>200 mi/h total</td>
<td></td>
</tr>
<tr>
<td>Tornado</td>
<td>160 mi/h rotational</td>
<td>NUREG-0800 (5)</td>
</tr>
<tr>
<td></td>
<td>40 mi/h translational</td>
<td>3.3.2 Tornado Loading</td>
</tr>
<tr>
<td>Volcanic ash</td>
<td>24 lb/ft² ground ash load</td>
<td>NUREG-0800 (5)</td>
</tr>
<tr>
<td></td>
<td>Site drainage basin: 7.4 in. for 6-hour probable maximum precipitation</td>
<td>3.8.4 Other Seismic Category Structures</td>
</tr>
<tr>
<td></td>
<td>Site drainage: 9.2 in. for 6-hour probable maximum precipitation</td>
<td></td>
</tr>
<tr>
<td>Flooding</td>
<td>Dry site for river flooding</td>
<td>ANSI/ANS-2.8-1992 (6)</td>
</tr>
<tr>
<td></td>
<td>Site drainage basin: 7.4 in. for 6-hour probable maximum precipitation</td>
<td>NUREG-0800 (5)</td>
</tr>
<tr>
<td></td>
<td>Site drainage basin: 9.2 in. for 6-hour probable maximum precipitation</td>
<td>2.4.2 Floods</td>
</tr>
<tr>
<td>Lightning</td>
<td>Lightning protection shall be provided for facility</td>
<td>NFPA 780 (7)</td>
</tr>
<tr>
<td>Snow</td>
<td>20 lb/ft² ground load</td>
<td>ASCE 7-93 (4)</td>
</tr>
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</table>


ANS = American Nuclear Society.
ANSI = American National Standards Institute.
ASCE = American Society of Civil Engineers.
DOE = U.S. Department of Energy.
NRC = U.S. Nuclear Regulatory Commission.
The major contributor to the frequency of aircraft impact into the ISA is general aviation aircraft during in-flight operations. The overall frequency of aircraft impact from all sources is $3.55 \times 10^{-4}$/yr (HNF-1786). DOE-STD-3014-96, *Accident Analysis for Aircraft Crash into Hazardous Facilities*, specifies that if the total impact frequency, calculated according to the method it gives, is less than $10^{-4}$/yr conservatively calculated, or $10^{-7}$/yr realistically calculated, the safety risk is below the level of concern and no design basis accident analysis is required.

Information on rotary wing aircraft is provided in Section 1.6.1 of the SNF Project FSAR. Medical evacuation helicopters’ closest approach to the ISA will be a landing pad to the west of the 2704 Building, which is about 0.9 mi from the ISA.

### D1.6.2 Other Transportation Accidents

A discussion regarding guidance for evaluating transportation accidents is provided in Section 1.6.2 of the SNF Project FSAR.

Highway 240 is 5 mi from the ISA. The nearest railroad not controlled by DOE is at the 1100 Area, located approximately 20 mi south of the ISA. DOE currently owns and controls the railroad north of this point, as discussed in Section 1.3.1 of the SNF Project FSAR. At these distances, explosive shipment on roads and railroads not controlled by DOE do not represent a threat to the ISA.

The main roadway and railroad controlled by DOE that pass nearest to the ISA are Route 4, which passes 1,155 ft to the west, and the mainline of the Hanford Railroad, which passes 9,000 ft to the north. The mainline of the Hanford Railroad is outside the above-listed safe distance, but Route 4 is not.

Regulatory Guide 1.91, * Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants*, allows for a risk assessment when the safe distance criterion is not met. If the exposure rate is less than $10^{-7}$ events/yr by best estimate analysis or less than $10^{4}$ events/yr by conservative analysis, then the risk of damage being caused by explosion is sufficiently low. The exposure rate, $r$, is defined as follows:

$$ r = n \times f \times s $$

where

- $n =$ explosion rate for the substance and transportation mode in question, per kilometer
- $f =$ frequency of shipment for the substance in question in shipments per year
- $s =$ exposure distance in km. This is the length of the roadway that would be within the safe distance of any portion of the facility. It is a function of the building dimension.
parallel to the roadway, the setback of the building from the roadway, and the calculated safe distance, R.

A recent study (H&R 522-1) concludes that a Hanford Site truck accident rate of $5.5 \times 10^8$ accident/mi should be used for risk analysis. This study was performed for shipment of radioactive materials and took credit for risk reduction factors that would be associated with such shipments. These included driver participation in special safety programs, the road conditions north of the Hanford Site Wye Barricade, and making shipments during off-peak traffic hours. A risk reduction factor of 40 was obtained for Hanford Site two-lane roads such as Route 4 (H&R 522-1). This risk reduction factor also would be appropriate for shipment of large quantities of explosive materials. For this analysis, it is conservatively assumed that the conditional probability that an explosion will result as a consequence of an accident is 0.1. Therefore, the explosion rate, $n$, is $5.5 \times 10^8$ explosions/mi.

There are currently no routine shipments of explosives on Route 4 of a quantity that would place the ISA at risk. However, such shipments could occur during the facility’s lifetime. For this analysis, the number of such shipments per year, $f$, was conservatively assumed to be six (H&R 522-1). The exposure distance, $s$, for the ISA was calculated to be 0.51 mi based on a building length of 257 ft, a setback from Route 4 of 1,155 ft, and a safe distance of 1,683 ft.

With these data, the exposure rate, $r$, was conservatively calculated to be $4.36 \times 10^8$ events/yr. As this is less than $10^6$ events/yr, no additional analysis is required.

Based on the above analyses, it is concluded that explosive shipments on roadways and railways, controlled and not controlled by DOE, do not represent a risk to the ISA.

### D1.7 NEARBY FACILITIES

Accidents in certain facilities in the 200 East Area (Figure D1-3) have the potential to impact the ISA facility and its operations, as discussed in Section D1.7.1. Conversely, certain nearby facilities can potentially be affected by accidents in the ISA, as discussed in Section D1.7.2.

Threats to the ISA from nearby facilities that are not known at this time will be subsequently evaluated by the unreviewed safety question process. This also applies to those activities that have been identified but that may change significantly in terms of a potential increased risk to the facility.
**D1.7.1 Potential Effects from Nearby Facilities**

Potential hazards to the ISA from on-site or off-site hazardous operations or facilities are examined under three general classifications:

- Non-reactor nuclear and non-nuclear industrial facilities within 5 mi of the ISA, including all activities conducted in and near the 200 East Area
- Nuclear reactors within a 50-mi radius of the ISA
- Military activities.

**D1.7.1.1 Hazards to the Interim Storage Area from Non- Reactor Nuclear Facilities.**

Facilities currently operating, recently operating, or with potential to operate at the 200 East and 200 West Areas were screened along with the area between the 200 East and 200 West Areas. For the ISA FSAR Annex, selected facilities were those believed to pose the most risk to safe operations at the ISA. Safety analysis reports and accident analyses prepared for these facilities were reviewed to determine possible hazards (e.g., radiological doses to personnel resulting from direct radiation, release of airborne radioactivity, or exposure to toxic chemicals).

Considered, but not included in this FSAR Annex, were the 200 East Area Burial Grounds, the Liquid Effluent Retention Facility, and the 200 Areas Effluent Treatment Facility in the 200 East Area. In the 200 West Area, the T Plant, U-Plant, Reduction-Oxidation (REDOX) Plant, and the 222-S Laboratory were considered, but not included. These facilities have insufficient radiological or toxicological inventories in a dispersible form to represent a risk to the ISA operation. The U-Plant and REDOX facilities have been shutdown for many years and are awaiting decontamination and decommissioning.

The specific facilities discussed here include the CSB, Plutonium-Uranium Extraction (PUREX) Facility, the Grout Treatment Facility, B Plant, the Waste Encapsulation and Storage Facility (WESF), the Tank Waste Remediation System (TWRS) facilities, 242-A Evaporator/Crystallizer, Plutonium Finishing Plant (PFP), and the Low-Level Waste Disposal Site. The worst-case scenarios for each of these facilities may challenge ISA habitability. Requirements for safely performing response actions are addressed in the CSB emergency response procedures. It should be noted that the following discussions address the worst-case scenarios and that safety-class and safety-significant SSCs and administrative controls have been provided at each facility to prevent or mitigate these potential accidents. During much of the ISA’s 40-year life, the facility will not be particularly sensitive to these hazards, as it will be simply a storage facility and will require little human presence.

**Canister Storage Building.** The CSB is located 0.25 mi. west of the ISA facility. Hazards from the CSB are described in detail in the SNF Project FSAR Chapter 3.0 and Annex A, Section A3.3. The analysis identifies hazard sources, hazardous conditions, potential accident scenarios and their initiators, and preliminary assessments of event frequencies and
consequences. Hazards are identified by form and location and represent a complete spectrum of events that could occur throughout the facility and have an impact on the ISA. The CSB has been assigned a final designation of hazard category 2 facility based on material at risk.

Six potential design basis accidents were analyzed for the CSB, as follows:

- Rearrangement of Multi-Canister Overpack (MCO) internals
- Gaseous release from the MCO
- MCO internal hydrogen explosion
- MCO external hydrogen explosion
- Thermal runaway reactions inside the MCO
- Violations of design temperature criteria.

The maximum 24-hour unmitigated 100 m dose (from the CSB) resulting from these design basis accidents was determined to be 300 rem and has the potential to require the evacuation of any personnel at the ISA facility.

**Plutonium-Uranium Extraction Facility.** The PUREX facility is located in the 200 East Area, 1.5 mi east-southeast of the ISA. It is the most recently constructed of the irradiated fuel separation facilities and was used for processing N Reactor fuel. The principal product was a solution of plutonium nitrate that was transferred to the PFP for further processing. Another product was uranyl nitrate solution, which was processed at the Uranium Trioxide Plant. The facility has been cleaned out to the extent practical and has been transitioned to surveillance and maintenance mode awaiting a final decision on disposal of the facility. The processing canyon area and the tunnels still contain loose contamination from previous processing activities. If the status of PUREX changes (i.e., active decontamination and decommissioning), potential impacts to the ISA will be re-evaluated.

The bounding accident for the PUREX facility in the surveillance and maintenance mode is a seismic event that releases a portion of the residual contamination in the ventilation gallery above the canyon. This accident would not impair the ability of personnel at the ISA facility to perform required safety actions because the quantity of material available for release is small.

**Grout Treatment Facility.** The Grout Treatment Facility is located on the eastern perimeter of the 200 East Area, approximately 2.7 mi east of the ISA. The Grout Treatment Facility combined tank wastes with grout-forming solids to form a grout slurry. The waste feed stream constituent of this slurry consisted of low-level fractions of radioactive wastes. The slurry was pumped into near-surface, concrete-lined vaults for permanent disposal. Only one vault has been filled with grout (completed in 1987) even though additional vaults have subsequently been
designed and constructed. This facility is currently not operational but has the potential to operate again in the future.

If the Grout Treatment Facility resumes operation, the maximum credible accident release postulated involves a double-ended jumper leak on the grout feed line, with the pit cover blocks left off. This postulated release would spray low-level waste as an aerosol for a maximum period of 24 hours before detection by a visual inspection. If the status of the Grout Treatment Facility changes (i.e., begins operations), potential impacts to the ISA will be re-evaluated. If the ISA habitability is challenged by an event at the Grout Treatment Facility, the appropriate action (i.e., notification, take-cover, emergency shutdown, evacuation) will be initiated using CSB emergency response procedures.

B Plant Facility. The B Plant is located approximately 1,500 ft east of the ISA. Until 1952, B Plant was operated as a fuel separation facility. In 1968, it was converted to a waste fractionation plant to remove $^{137}$Cs and $^{90}$Sr from radioactive waste streams. This had the effect of reducing heat loads in the double-shell tanks. The B Plant currently is deactivated and has been transitioned to surveillance and maintenance mode.

The worst-case credible accident at B Plant is a postulated flooding of the 291-B high-efficiency particulate air filters and a subsequent hydrogen explosion. The dose to the collocated worker at 100 m from B Plant was calculated to be 952 rem, and the maximum off-site dose to be 0.368 rem at the Columbia River, 16.8 mi east of B Plant (HNF-SD-WM-BIO-003). Systems and administrative controls are in place at B Plant to mitigate the releases. The worst-case credible accident at the B Plant involves the filters, until they are decontaminated and decommissioned. This scenario has the potential to require evacuation of any personnel at the ISA facility. CSB emergency response procedures will describe the action to be taken by personnel at the ISA facility in the event take-cover or evacuation actions are required by events at nearby facilities.

Waste Encapsulation and Storage Facility. The WESF is distinct from B Plant, even though it is located on the west end of B Plant and shares a common wall with the plant. Historically, WESF was involved in converting the cesium and strontium removed from waste streams at B Plant into cesium chloride and strontium fluoride salts. These materials were then encapsulated in double-walled metal containers and stored in a water-filled cooling basin. Strontium fluoride and cesium chloride capsules are still being stored in this fashion at WESF, but no new capsules are being produced.

The worst-case credible accident postulated at WESF involves a loss of cooling water in the storage pool, with a subsequent rupture of capsules and aerosolization of the capsule contents (HNF-SD-WM-BIO-002). The maximum dose to the public receptor at the near bank of the Columbia River, about 18 miles from WESF, is calculated to be 9 rem. The 24-hour unmitigated dose at 100 m is 1,700 rem from inhalation and 32 rem from radiation shine. Systems and administrative controls are in place at WESF to mitigate the releases. However, this scenario has the potential to require evacuation of ISA operating personnel. CSB emergency response
Annex D - 200 Area Interim Storage Area

procedures will describe the action to be taken by personnel at the ISA facility in the event take-cover or evacuation actions are required by events at nearby facilities.

**Tank Waste Remediation System.** The TWRS facilities include 149 single-shell tanks and 28 double-shell tanks for the storage of liquid radioactive mixed waste solutions from the Hanford Site's chemical processing plants and associated support facilities, which include holding tanks, transfer lines, and valve pits. The TWRS facilities are located in both the 200 East and 200 West Areas.

The TWRS facilities nearest to the ISA are the B, BX, and BY Tank Farms, located approximately 0.8 mi northeast of the ISA. Each of the farms has 12 single-shell tanks. The B and BX single-shell tanks are rated at 500,000 gal each, and those in the BY Tank Farm are rated at 750,000 gal each. In addition, there are four 55,000-gal tanks in the B Tank Farm, and a double-contained receiver tank serves the BX and BY Tank Farms. The tanks in these tank farms were constructed between 1943 and 1949.

The worst-case accident postulated for the TWRS facilities is the inadvertent pumping of liquid waste through an open valve into a valve pit and consequent overflow of the pit, which results in a dose of 330 rem at 100 m. The postulated cause of the accident is mis-routing of waste during transfer operations (HNF-SD-WM-BIO-001). Waste transfers into Hanford Site single-shell tanks are no longer allowed. Most of the tanks in these three farms have been interim stabilized, meaning that as much of the liquid has been removed from the waste as practicable, leaving solids with a minimum liquid content. Two tanks, 241-BY-105 and 241-BY-106, have not yet been interim stabilized, so pumping activities involving these tanks and double-contained receiver tank 244-BX are expected in the future (HNF-EP-0182-116). Administrative controls are in place at the tank farms for early detection and mitigation of a release caused by mis-transfer during waste transfer operations. Also considered was a flammable gas deflagration in a tank that could result in a dose of 650 rem at 100 m. However, because of the site distance (0.8 mi) and the flammability controls, this scenario would not significantly impact the ISA. Even though the cross-site transfer line passes near the ISA, a leak from the line is not considered a credible event because of its safety-class, encased-pipe design (HNF-SD-WM-BIO-001, Addendum 2).

If there was a release from a TWRS accident, the immediate action for personnel at the ISA facility would be to respond to the take-cover sirens. Depending on the specific conditions, follow-up actions could include evacuation per CSB emergency response procedures.

Eventual retrieval of the tank wastes for final disposal is planned for the future. Activities involving waste removal and decontamination and decommissioning of the tanks will be evaluated at that time for their effect on the ISA safety analysis.

**242-A Evaporator/Crystallizer.** The 242-A Evaporator/Crystallizer is located 1.5 mi east of the ISA. The 242-A Evaporator/Crystallizer uses evaporative concentration to reduce the volume of liquid wastes. The concentrated slurry, reduced in volume, is transferred and stored in
underground double-shell waste storage tanks. The process condensate is routed to the Liquid Effluent Retention Facility for storage and treatment at the Effluent Treatment Facility.

The worst-case accident scenario reported in the 242-A Evaporator/Crystallizer Safety Analysis Report (HNF-SD-WM-SAR-023) involved a release from a spray leak in the pump room and a failure of the exhaust system's high-efficiency particulate air filters. Although this event is considered extremely unlikely, the event was analyzed for safety-class determinations at the 242-A Evaporator/Crystallizer. The release from a spray leak is comprised of a liquid component and an aerosol component. The analysis assumed that the liquid component would remain in the pump room; however, the aerosol component would be released to the environment via the K1 exhaust stack. The maximum dose consequence to the on-site individual at 100 m from the evaporator would be 14 rem effective dose equivalent. This scenario might require action to prevent exposure of ISA operators to the release (i.e., take cover), but it is not likely to require shutdown of ISA operations. CSB emergency response procedures will describe the action to be taken by personnel at the ISA facility in the event take-cover or evacuation actions are required by events at nearby facilities.

Plutonium Finishing Plant. Located near the western boundary of the Hanford Site in the 200 West Area, the PFP is 5.0 mi west of the ISA. The PFP converts plutonium nitrate solution to plutonium oxide and performs plutonium handling and storage operations. Contaminated liquid waste streams from the PFP are routed to the tank farms. This facility is in transition from its previous special nuclear material processing mode to preparation for decontamination and decommissioning.

During its previous mission of processing special nuclear materials, plutonium-bearing materials were produced and still exist in the facility. These include pure and mixed oxides, fluorides, oxalates, silicates, and organic-based sludge and residues. The current PFP missions are defined as (1) receiving, sorting, storing, and shipping special nuclear material, (2) stabilizing the reactive materials that remain in the plant, (3) providing laboratory support for other Hanford Site facilities, (4) handling radioactive and mixed waste, and (5) shutdown facility surveillance (WHC-SD-CP-SAR-021). As part of the material stabilization activities, preparations are being made for startup of a calcining operation. Revisions to the safety analysis report to include this operation are in progress.

The bounding accident for the PFP is a seismic event with a horizontal ground acceleration of 0.2 g. The consequences for this event are calculated to be 15.2 rem effective dose equivalent at the nearest occupied facility in the worst-case direction, 1,800 ft west-northwest of PFP, and 0.31 rem at the Site boundary, 7.8 mi west. Because of the distance separating the ISA from the PFP, the release would not adversely affect ISA operations and would not require evacuation of personnel at the ISA facility.

Low-Level Waste Disposal Site. The commercial low-level waste disposal site operated by U.S. Ecology, Inc., is the only non-DOE industrial facility within 5 mi of the ISA (Figure D1-1). The disposal site is on land leased from Washington State located 1 mi southwest
of the ISA. The low-level waste is buried in U.S. Department of Transportation-approved shipping containers. Monitoring of groundwater and vegetation is performed as required by the facility’s NRC operating license and environmental impact statement. This facility is not likely to have a significant accidental airborne radioactive release that could adversely affect the ISA.

D1.7.1.2 Hazards to the Interim Storage Area from Non-Nuclear Hanford Site Facilities. A number of non-nuclear industrial facilities operating in the 200 Areas pose the potential for accidental fires, explosions, or releases of toxic fumes. These include the Essential Materials Warehouse (Building 275-EA), oil and paint storage buildings, fabrication shops, gas cylinder storage buildings, the spare parts and electrical warehouse, B Plant storage buildings, maintenance facilities, gasoline service stations, and the powerhouse complexes in each area (284-E and 284-W). Considering its location, Building 275-EA may have the potential to pose a risk to ISA operations and personnel and is detailed below. The 2,000-lb chlorine bottles previously located at Building 283-E have been removed and are no longer a hazard to the ISA.

Building 275-EA. Building 275-EA, the Essential Materials Warehouse, was constructed in 1955 and is located near the PUREX facility about 1.5 mi east-southeast of the ISA. It is classified as an unprotected wood frame structure and is susceptible to collapse as a result of an external event (e.g., earthquake, wind, snow, or ash loading) or an internal event (e.g., forklift collision with a bearing wall, or fire). Building 275-EA currently stores more than 100 different types of potentially hazardous solids and liquids including acids, bases, solvents, fluorides, pesticides, and herbicides. Radioactive materials are not stored in this building.

The worst-case chemical release postulated in the safety analysis occurs following a building collapse whereupon 2,450 kg of 1,1,1-trichloroethane evaporates under adverse atmospheric conditions. The maximum concentration at the PUREX facility gatehouse and environs is calculated to be 187 ppm, which is below the time-weighted average threshold limit value of 350 ppm (i.e., the concentration for this chemical that an industrial worker may be repeatedly exposed to without adverse effects) as given in Threshold Limit Values and Biological Exposure Indices for 1989-1990 (ACGIH 1989). Concentration levels at the ISA would be significantly less than this because of the 1.5 mi separation.

D1.7.1.3 Hazards to the Interim Storage Area from Nuclear Reactors. Three recently operating reactors, the N Reactor, Fast Flux Test Facility (FFTF), and the Critical Mass Laboratories, no longer pose a threat to the ISA. The N Reactor is undergoing decontamination and decommissioning, the FFTF is in standby mode and may operate again in the future, and the Critical Mass Laboratories in the 200 East Area north of the PUREX facility are currently being used as tank farm office areas.

The N Reactor was a 4,000 MW, dual-purpose, pressure tube, light-water cooled, graphite-moderated reactor (UNI-M-90). It is located in the 100 N Area and is about 5.5 mi from the nearest Hanford Site boundary and about 9 mi from the ISA. The N Reactor began operating in 1964 and produced plutonium for the defense program and steam for electrical power generation. The reactor was shut down in 1987 for safety improvements and then subsequently
defueled and placed in cold standby in 1988. It is currently being decontaminated and decommissioned. Current plans are to remove the reactor building structures for the production reactors down to the reactor block and then cocoon the reactor block for 75-year safe storage.

The FFTF is a 400 MW, sodium-cooled, mixed-oxide-fueled reactor that is currently in standby status with the fuel removed from the core. Sodium is kept circulating in the loops at 400 °F until a decision is made to either restart the reactor for future missions or decommission it. The FFTF is located in the 400 Area and is approximately 4.5 mi from the nearest Hanford Site boundary, which is to the east of the facility (Figure D1-2) and about 13.5 mi. from the ISA. Spent FFTF fuel, removed from storage in liquid sodium and packaged in double-walled casks with an inert cover gas, is stored in the ISA at the 400 Area.

The bounding accidents for the FFTF in its current status involve (1) a liquid sodium spill and (2) damage to a cask and its contents in the ISA (HEDL-TI-7500-FSAR). The sodium spill scenario postulates a spill of 180,000 lb of activated liquid sodium. The 2-hour dose at 1.5 mi from the FFTF is 0.015 mrem. The dose at 4.5 mi from the facility for a 30-day exposure is 0.26 mrem. The maximum credible cask release at the 400 Area ISA postulated cracking in 100% of the fuel pins of one cask, with crushing and exposure of 1% of the fuel material. The dose at 100 m from the facility was 4.5 rem, the maximum dose at the Site boundary was 4.0 mrem. Because of the distance from the 200 Area ISA, these accidents do not pose a hazard to ISA operations.

The only operating nuclear reactor on the Hanford Site is Washington Nuclear Plant (WNP)-2. The location of this reactor is shown in Figure D1-2. WNP-2 is an operating commercial nuclear power plant using a boiling-water reactor steam supply system. The design power level was increased to 3,486 MW in 1995 (Docket No. 50-397). The reactor was designed by the General Electric Company and is designated as a BWR/5 with a Mark II containment.

By the requirements of 10 CFR 100, “Reactor Site Criteria,” the following are the maximum allowable doses for WNP-2:

<table>
<thead>
<tr>
<th>Location</th>
<th>Duration</th>
<th>Whole body dose</th>
<th>Thyroid</th>
</tr>
</thead>
<tbody>
<tr>
<td>Exclusion area boundary</td>
<td>2 hours</td>
<td>250 mSv (25 rem)</td>
<td>3,000 mSv (300 rem)</td>
</tr>
<tr>
<td>Low population zone</td>
<td>30 days</td>
<td>250 mSv (25 rem)</td>
<td>3,000 mSv (300 rem)</td>
</tr>
</tbody>
</table>

The exclusion area boundary for WNP-2 is 1.2 mi and the low population zone distance is 3 mi. The ISA is located approximately 11 mi from WNP-2. Using the atmospheric diffusion guidance provided in NRC Regulatory Guide 1.3, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors, to estimate the dose reduction as a function of distance, it was determined that the 2-hour and
30-day doses at the ISA would be reduced by a factor of 20. Review of the site-specific meteorology provided in the WNP-2 FSAR (Docket No. 50-397) shows there would be no significant reduction for the change in wind direction. The factor of 20 reduction for distance would result in a whole body dose of 12.5 mSv (1.25 rem) and a thyroid dose of 150 mSv (15 rem). The expected dose that would be received at the ISA if a loss of coolant accident occurs would be significantly less than this. NRC Regulatory Guide 1.3 requires an assumption that 25% of the radioactive iodine and all of the noble gases are released to the containment. In fact, the emergency core cooling system would prevent most of these releases, as little fuel damage would occur as a result of the loss of coolant accident.

Energy Northwest (formerly known as Washington Public Power Supply System) plans to add an independent spent fuel storage installation on their leased property. The independent spent fuel storage installation will be licensed following the requirements in 10 CFR 72. According to 10 CFR 72.106, an individual located at the installation’s controlled area boundary shall not receive a dose greater than 5 rem. At the ISA, this would result in a dose not exceeding 0.25 rem. CSB emergency response procedures will describe the action to be taken by personnel at the ISA facility in the event take-cover or evacuation actions are required by events at nearby facilities.

**D1.7.1.4 Hazards to the Interim Storage Area from Industrial Facilities Off the Hanford Site.** There are no oil or gas pipelines in the vicinity of the ISA. The nearest major natural gas pipeline to the ISA site is about 29 mi. A 20-in. gas transmission line of the Northwest Pipeline Corporation is located east and essentially parallel to U.S. Highway 395 between Pasco and Ritzville, Washington. A second pipeline system consisting of parallel 36-in. and 42-in. lines, owned by Pacific Gas Transmission Company, passes through Wallula, approximately 33 mi from the site (Hosler 1996). These distances eliminate any potential hazard to plant operations from a natural gas fire or explosion.

The nearest petroleum product storage tanks are located 38 mi from the site. These tanks include 23-million-gallon capacity tanks at the Chevron Pipeline Company and 21-million-gallon capacity tanks at the Tidewater Barge Lines. There are no plans to use a third petroleum storage facility at the Port of Pasco (Hosler 1996).

Located within the Richland city limits is the Siemens Power Corporation’s Richland Engineering and Manufacturing Facility. All operational steps for the manufacture of nuclear fuel for light water reactors are conducted within the facility, including the conversion of UF₆ to UO₂. The most limiting postulated accident at this facility is a fire in the UF₆ cylinder storage area (Siemens 1994). Fusible plugs in twelve cylinders are assumed to melt causing the release of UF₆, which reacts with moisture in the air to form UO₂F₂ (solids in the form of uranyl fluoride hydrates) and 4HF (as hydrogen fluoride gas). UF₆ is a radiological hazard by inhalation. However, UF₆ is also a concern because of its chemical toxicity and the associated HF that can cause skin and eye burns and lung impairment. This accident results in a dose that exceeds 10 mSv (1 rem) out to 1.2 mi. The accident also exceeds the Emergency Response Planning Guide-2 (ERPG-2) toxicology limits out to 8.8 mi (ACGIH 1989). The ERPG-2 is the maximum...
The airborne concentration below which it is believed that 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 their abilities to take protective action. The ISA is located 19 mi from the Siemens facility. Therefore, operators would not be placed at risk by an accident at the Siemens facility.

No other non-nuclear industrial facilities or operations have been identified that can impact ISA operations.

D1.7.1.5 Hazards to the Interim Storage Area from Military Facilities. The Yakima Training Center is a sub-installation under the command of Fort Lewis (Tacoma, Washington). Further information is given in the *Final Environmental Impact Statement — Ft. Lewis Military Installation* (DOA 1979). The southeastern boundary of the Yakima Training Center is located about 23 km (14 mi) from the ISA (see Figure D1-12). The Yakima Training Center is used for military maneuvers and weapons training and is the only significant military activity in the vicinity of the Hanford Site.

The only weapon currently in use at the Yakima Training Center known to present a hazard to the Hanford Site is the Multiple Launch Rocket System (MLRS). With a range of approximately 16 mi, the MLRS could potentially impact the ISA site. However, the MLRS is only fired from the perimeter of the Yakima Training Center into a centrally located impact zone. The safety fan for the MLRS is shown in Figure D1-12. The MLRS is fired away from the Hanford Site and is only fired with dummy warheads. Given this information, additional safety features, and the administrative controls in place at the Yakima Training Center, a weapons accident having an impact on the Hanford Site is very improbable.

A more probable hazard to Hanford Site facilities is a scenario in which a fire started within the Yakima Training Center boundary spreads to the Hanford Site. Exploding artillery shells, sparks from tracked vehicles or other machines, and careless smoking by troops might start brush fires that, under adverse meteorological conditions, could spread rapidly beyond the Yakima Training Center boundaries. The hazards associated with range fires are discussed in Chapter 3.0 of the SNF Project FSAR.

D1.7.2 Potential Effects to Nearby Facilities

ISA accidents with the potential to affect the maximum on-site individual (which may include persons at some of the facilities discussed above) are discussed in Section D3.4.2. Section 15.4.2.2 of the SNF Project FSAR requires that a hazards assessment for each SNF Project facility be prepared based on the facility-specific hazards and safety analyses that are contained in each SNF Project Annex. The scope of the hazards assessment is provided in Section 15.4.2.2 of the SNF Project FSAR. The hazards assessment characterizes the potential consequences on workers, the public, and the environment for each postulated accident and determines the emergency planning zone for each facility as well as the emergency class.
protective actions, and the observable indications and criteria (emergency action levels) corresponding to the range of identified accidents. The hazards assessment provides the framework for response to virtually any declared emergency.

As stated in Section 15.4.3 of the SNF Project FSAR, prompt and accurate emergency notifications would be made to mitigate consequences and to protect the health and safety of workers, the public, and the environment in accordance with DOE/RL-94-02, Hanford Emergency Management Plan. The Emergency Operations Center (EOC) is responsible for notification of affected areas and other contractors on-site. The DOE Richland Operations Office EOC is responsible for follow-up notifications when emergencies are reclassified or terminated in accordance with Section 15.4.3.4 of the SNF Project FSAR.

D1.8 VALIDITY OF EXISTING ENVIRONMENTAL ANALYSES

No significant discrepancies have been identified between the site characteristic assumptions made in this chapter and those made in the site-wide SNF Project Environmental Assessment (DOE/EIA-1185).

D1.9 REFERENCES


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Figure D1-1. Location of the 200 Area Interim Storage Area Relative to the Canister Storage Building.
Figure D1-2. Hanford Site Boundaries.
Figure D1-3. On-Site Population Distribution in the 200 East Area by Zone.
Figure D1-4. Wind Rose for the 200 East Area.

HMS Data from 1983 to 1991
(Rings are in 2% increments)
Figure D1-5. Wind Speed Histogram for the 200 East Area.
Figure D1-6. Hanford Site Topographic Map and Cross Section.
Figure D1-7. Geology of the 200 East and West Areas.
Figure D1-8. Stratigraphy of the 200 East Area.
Figure D1-9. Log of Borings: Boring VP-15.

Key to Log Borings

32 Blows required to sampler one foot

■ Indicates depth at which undisturbed sample was extracted

☑ Indicates depth at which disturbed sample was extracted

Boring completed at depth of 62.5 ft on 8/2/89
No groundwater was encountered during drilling
Hammer weight = 750 lb.
Hammer drop = 18 in.

Depth in Feet
Sample Symbol

Depth in Feet
Sample Symbol

Elevation 709 +/-

Light brown silt and very fine to fine sand (loose to medium dense)

Light brown silty fine sand with some gravel and cobbles (medium dense)

Light brown or grayish brown fine to medium sand, trace of coarse sand, trace of silt, and occasional gravel (dense to very dense)
Figure D1-10. Geologic Cross-Section of the Canister Storage Building Site.
Figure D1-11. Dynamic Soil Properties of the Canister Storage Building Site.

<table>
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<tr>
<th>Depth (ft)</th>
<th>Natural Density (lb/ft³)</th>
<th>Compressional Wave Velocity (ft/s)</th>
<th>Shear Wave Velocity (ft/s)</th>
<th>Dynamic Shear Modulus (lb/ft²)</th>
<th>Dynamic Poisson's Ratio</th>
<th>Elevation (ft)</th>
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<td>300</td>
</tr>
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* Estimated using data from previous investigations.
Figure D1-12. The Location of Yakima Training Center with Respect to the Hanford Site.
CHAPTER D2.0

FACILITY DESCRIPTION
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The 200 East Area Interim Storage Area (200 Area ISA) at the Hanford Site provides for the interim storage of non-defense reactor spent nuclear fuel (SNF) housed in dry cask storage systems. The 200 Area ISA is a relatively simple facility consisting of a boundary fence with gates, lighting along the perimeter, and pads on which to place dry storage casks. The pads are concrete and level gravel placed by the project based on HNF-2524, 200 East Area Interim Storage Area Preliminary Safety Evaluation, approved Site Evaluation Form (#2E-96-10), and an "Approval to Construct" authorization (Hanson 1998). The fence supports safeguards and security and provides a radiation protection buffer zone.

An equipment storage building (pre-engineered metal building) is proposed near the 200 Area ISA to house lifting devices, impact limiters, transfer casks, and various other types of equipment associated with storage, movement, or transport of the casks. This storage building will be located outside the 200 Area ISA and provides no safety-related function.

The three different dry cask storage systems used at the 200 Area ISA are as follows:

- Interim Storage Cask (ISC) used for the Fast Flux Test Facility (FFTF) SNF
- Neutron Radiography Facility (NRF) Training, Research and Isotope Production, General Atomics (TRIGA\(^1\)) casks and U.S. Department of Transportation (DOT)-6M containers within a Chem-Nuclear Services, Incorporated, Rad-Vault\(^2\) storage vault used for NRF TRIGA SNF
- Nuclear Assurance Corporation (NAC)-I casks within International Standards Organization (ISO) containers used for commercial light water reactor (LWR) SNF from the 300 Area.

The 200 Area ISA is located approximately 0.25 mi west of the Canister Storage Building (CSB). The footprint of the ISA is nominally 500 ft. by 400 ft. surrounded by a chain-link fence, with gates in the fence that control access of vehicles and personnel. Light poles provide illumination for the 200 Area ISA. Within the fenced area are concrete pads for placement of the ISC and the NAC-I casks within the ISO containers. The Rad-Vault holding NRF TRIGA casks and DOT-6M containers is to be placed on a level gravel pad placed by the project.

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\(^1\) TRIGA is a registered trademark of Gulf General Atomics Company, Inc.

\(^2\) Rad-Vault is a trademark of Chem-Nuclear Systems, Inc.
Interim storage at the ISA is intended until shipment of the materials to a disposal facility. Loaded containers are to be stored at the ISA for a period of up to 40 years.

D2.1 INTRODUCTION

This chapter provides information that satisfies the safety analysis report requirements of U.S. Department of Energy (DOE) Order 5480.23, Nuclear Safety Analysis Reports, paragraphs 8.b.(3)(d), as amplified in Attachment 1, paragraphs 4.f. (3) (d) 4 a, and 4.f. (8) (b) and (c)2 of the Order, and DOE-STD-3009-94, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports. Facility and component descriptions are provided in support of assumptions used in the hazard analysis and accident analysis discussed in Chapter D3.0. This chapter provides summary level facility and storage system descriptions. Additional information at the level of functional requirements and performance evaluation of the safety structures, systems, and components (SSCs) is provided in Chapter D4.0.

Consistent with DOE-STD-3009-94 guidance, descriptions in this chapter provide a model of the facility systems necessary to develop an understanding of the structures, equipment, and operations without extensive consultation of more controlled design references.

The following topics are discussed in this chapter to the level of detail specified by the precepts of the graded approach described in DOE Order 5480.23:

1. Overview of the facility (Section D2.3)
2. Description of the facility structure and design basis (Section D2.4)
3. Description of the facility process systems and components, and relationships of the SSCs (Section D2.5)
4. Description of confinement systems (Section D2.6)
5. Description of facility safety support systems (Section D2.7)
6. Description of facility electrical systems (Section D2.8)
7. Description of facility auxiliary systems and support facilities (Section D2.9).
D2.2 REQUIREMENTS

D2.2.1 U.S. Department of Energy Regulations, Orders, and Standards

The following DOE Orders, regulations, and standards are applicable to the safety basis of the facility:

- Title 10, Code of Federal Regulations (CFR), Section 830.120, "Quality Assurance Requirements." This rule requires that a sufficient quality assurance program be in place.

- 10 CFR 835, "Occupational Radiation Protection." This rule provides requirements for radiation protection programs.

- DOE Order 5480.23, Nuclear Safety Analysis Reports. This order provides nuclear safety analysis report content requirements.

- DOE Order 6430.1A, General Design Criteria. This order for nonreactor nuclear facilities presents the main reference standards and guides for facility design. In addition, Division 13, "Special Facilities," Section 1300, "General Requirements," and Section 1320, "Irradiated Fissile Material Storage Facilities," requirements are imposed for the 200 Area ISA.

- DOE-STD-1027-92, Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports. This standard is used to determine hazard categories for nuclear facilities.

- DOE Order 5480.28, Natural Phenomena Hazards Mitigation. This order is used to define design requirements for seismic events and straight wind.

- DOE-STD-1020-94, Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities. This standard provides natural phenomena hazard design and evaluation criteria. The 200 Area ISA was designed and evaluated for seismic events and straight wind in accordance with this standard.

- DOE-STD-1021-93, Natural Phenomena Hazards Performance Categorization Guidelines for Structures, Systems, and Components. This standard is used to define the specific performance category for SSCs.
D2.2.2 U.S. Nuclear Regulatory Commission Requirements and Guidance

In Letter 95-SFD-167, Implementation of K Basins Spent Nuclear Fuel Project (SNFP) Regulatory Policy (Sellers 1995), DOE established the requirement for new SNF Project facilities to achieve "nuclear safety equivalency" to comparable U.S. Nuclear Regulatory Commission (NRC)-licensed facilities. The SNF Project identified the NRC requirements that were needed in addition to existing and applicable DOE requirements to establish nuclear safety equivalency. These NRC requirements and the process used to identify them are documented in HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements, and in WHC-SD-SNF-DB-009, Canister Storage Building Natural Phenomenon Hazards. 10 CFR 72.3, "Definitions," defines SSCs that are considered "important to safety." 10 CFR 72.122, "Overall Requirements," requires that the design bases for SSCs important to safety reflect appropriate combinations of effects on normal and accident conditions and the effects of natural phenomena.

For the 200 Area ISA, important-to-safety SSCs have been identified based on the analysis performed in Chapter D3.0 and in accordance with 10 CFR 72.3. Once SSCs have been identified as having a function meeting the definition of important to safety, the requirements for SSCs important to safety specified in 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," are imposed. A graded approach is applied to an SSC important to safety by using the guidance provided in NUREG/CR-6407, Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety. This classification process is described in more detail in Section D4.2.

D2.2.3 Other Codes and Standards

Appendix A of HNF-SD-SNF-PMP-218, Site-Wide Spent Nuclear Fuel Project Management Plan, summarizes the requirements of the 200 Area ISA. The 200 Area ISA Standards/Requirements Identification Document identifies federal, state, and local regulations and laws with applicability to the 200 Area ISA. These requirements provide the basis for the design, development, and testing of equipment to safely store SNF at the 200 Area ISA.

DOE/RL-92-36, Hanford Site Hoisting and Rigging Manual, provides the requirements for lifting and rigging equipment. All lifts of SNF or packages containing SNF are considered critical lifts.

The documents identified in this section establish the design requirements for the 200 Area ISA.
D2.3 FACILITY OVERVIEW

The ISA is located within the Hanford Site 200 East Area. Its purpose is to provide the location for aboveground dry cask storage of SNF. The 200 Area ISA will provide safe outside storage of the SNF, while protecting fuel integrity through the use of storage systems resistant to natural phenomena hazards. While the majority of the fuel to be stored within the ISA will consist of FFTF SNF, the ISA will also store other SNF from the Hanford Site, including NRF TRIGA and Material Characterization Center (MCC) commercial LWR fuel.

An equipment storage building (pre-engineered metal building) is proposed near the 200 Area ISA to house lifting devices, impact limiters, transfer casks, and various other types of equipment associated with storage, movement, or transport of casks. This storage building will be located outside the 200 Area ISA fence and provides no safety-related function.

An aboveground dry cask storage location (the ISA) is necessary for the spent fuel because the current storage facilities are being shut down and deactivated. The spent fuel will be transferred to interim storage because there is no permanent repository storage currently available.

The 200 Area ISA is located west of the CSB. The footprint of the ISA is nominally 500 ft. by 400 ft. surrounded by a 7-ft. tall chain-link fence topped with three strands of barbed wire. Five manual gates in the fence, four 30-ft sliding and 1 personnel gate, control access of vehicles and personnel to the 200 Area ISA. Light poles within the perimeter provide illumination. Within the fenced area are three concrete pads; two for placement of ISCs and one for the NAC-1 casks within ISO containers. The Rad-Vault holding NRF TRIGA casks and DOT-6M containers will be placed on graded, compacted gravel.

The two pads for ISC storage are approximately 171 ft. by 26 ft. by 1.5 ft. thick constructed of reinforced concrete. The NAC-1 pad is approximately 88 ft. by 28 ft. by 1 ft. thick constructed of reinforced concrete. This smaller pad runs parallel to the two larger ones.

The three pads are laid out along their length on the north-south axis. The two ISC pads are separated nominally by 43 ft., and the ISO/NAC-1 pad is separated by 80 ft. from the nearest ISC pad.

The level gravel area for the Rad-Vault is located ~30 ft. northwest of the NAC-1 storage pad. A representation of the facility and its relation to the CSB are provided in Figure D2-1.

The 200 Area ISA facility components (excluding the dry cask storage systems) are classified as General Service; the dry cask storage systems are designated Safety Significant. The core component container (CCC) and the NAC-1 canister are designated as Safety Class and have both safety-class and safety-significant functions.
The ISA is surrounded by a galvanized steel chain-link fence at least 7 ft. high to restrict personnel access. Access to the facility is provided via four manually operated truck gates and one personnel gate. 10 CFR 72, Sections 72.180, 72.182, 72.184, and 72.186 require physical security plans and system designs in accordance with 10 CFR 73, "Physical Protection of Plants and Materials." Security aspects are not addressed in this final safety analysis report (FSAR).

Sodium lamps mounted on poles located inside the 200 Area ISA at the fence line are provided for illumination of the facility. There are eight light poles located two per side of the fenced enclosure.

No uncontained radioactive materials will be handled at the storage area. Therefore, decontamination and decommissioning efforts should be minimal.

There is interaction of the ISA with existing CSB facilities for surveillance activities and central alarm notification.

Table D2-1 summarizes the specific casks and containers to be used at the 200 Area ISA. The NAC-1 cask generically refers to either the NAC-1 or Nuclear Fuel Services (NFS)-4 SNF shipping casks that will satisfy both on-site transportation and storage requirements. The NAC-1 and NFS-4 casks were fabricated to the same design drawing, but at different times by different corporate owners. Both model casks will be referred to as the NAC-1 cask throughout this document.

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<tr>
<th>Cask/Container</th>
<th>Weight</th>
<th>Dimensions</th>
</tr>
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<tbody>
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<td>114,200 lbs. (max)</td>
<td>85 in. diameter x 181 in. height</td>
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<td>Rad-Vault</td>
<td>81,400 lbs. (max)</td>
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<td>TRIGA</td>
<td>2,013 lbs. (loaded)</td>
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<tr>
<td>DOT-6M</td>
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<td>23 in. diameter x 70 in. height</td>
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<tr>
<td>ISO</td>
<td>8,624 lbs. (empty)</td>
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<tr>
<td></td>
<td>4,400 lbs. (empty)</td>
<td>6 ft. high x 8 ft. wide x 20 ft. long</td>
</tr>
<tr>
<td>NAC-1</td>
<td>47,150 lbs. (loaded)</td>
<td>50 in. diameter (max) x 214 in. long</td>
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</tbody>
</table>

DOT = U.S. Department of Transportation.
ISC = interim storage cask.
ISO = International Standards Organization.
NAC = Nuclear Assurance Corporation.
TRIGA = Training, Research and Isotope Production, General Atomics.
D2.3.1 200 Area Interim Storage Area Hazards

A final hazard category classification of the facility is presented in Chapter D3.0, with the 200 Area ISA classified as a Hazard Category 2 facility. This classification was made in accordance with DOE-STD-1027-92.

The 200 Area ISA hazard analysis is documented in SNF-4820, *200 Area Interim Storage Area Final Hazard Analysis Report*. The only significant inventory of hazardous material in the 200 Area ISA is the radiological content of the dry cask storage systems. The 200 Area ISA does not house nor will it routinely conduct any chemical processes. Radiological guidelines applicable to the radionuclide inventory were found to be more limiting than the toxicological guidelines for the release of SNF particulate. Other hazardous materials identified in the dry cask storage systems include potentially pyrophoric metals and hydrides, oxidizers, and hydrogen.

The results of a fire hazards analysis (SNF-4932) concluded that no release of radioactive material to the environment resulted from any facility fire scenario.

D2.3.2 200 Area Interim Storage Area Operations Summary

200 Area ISA operators, supervisors, and managers will be provided by CSB staff.

Placement of Fast Flux Test Facility Fuel Containers

Receipt and set up of the mobile crane precedes the arrival of the transporter carrying the ISC. After the tie-downs are removed from the ISC, the lifting fixture will be attached to the mobile crane, as shown in Figure D2-2. Once the ISC lifting fixture is securely attached to the ISC lifting lugs, the ISC will be hoisted and removed from the transporter. The ISC will be placed in its assigned storage position on the concrete pad. The environmental cover on top of the ISC will be installed. The transporter used to carry the ISC can then be released.

Placement of the NRF TRIGA Cask and DOT-6M Containers in the Rad-Vault

The Rad-Vault, Figure D2-3, is not designed to be moved in a loaded condition or with the lid installed. Therefore, loading of the TRIGA casks (see Figure D2-4 and D2-5) and DOT-6M containers (see Figure D2-6) into the Rad-Vault will take place in the 200 Area ISA. Each Rad-Vault is capable of receiving two DOT-6M containers and six TRIGA casks. The casks and the containers are transported two at a time to the 200 Area ISA. Only a single cask or container is allowed to be moved into the Rad-Vault at a time. The TRIGA cask and the DOT-6M container have no specified loading order. Specific details, drawings, and operational restrictions for cask movement are provided in separate procedures. Empty 55-gallon drums are used as spacers inside the Rad-Vault and swapped out one at a time as loaded fuel casks are placed into the Rad-Vault. Final loading will have six TRIGA casks, two DOT-6M containers,
and one spacer drum (located in the center position), as shown in Figure D2-7. The spacer drums are only used for ease of loading and provide a passive function of providing a close packed array such that the containers cannot tip over during or after placement.

Prior to the receipt of the transporter carrying the Rad-Vault body, the mobile crane will be positioned and set up. Once the transporter carrying the Rad-Vault lid has been received, the Rad-Vault body can be placed at the assigned storage location. The Rad-Vault lid is placed adjacent to the Rad-Vault body. The transporters used to carry the Rad-Vault and lid can then be released.

Reception of the transporter carrying two DOT-6M containers can proceed. Using a 55-gallon drum lifter, a spacer drum will be sequentially removed from the Rad-Vault and replaced with a loaded DOT-6M container. This process will be repeated for the second DOT-6M container. The transporter that carried the DOT-6M containers can then be released.

A similar substitution process is used for loading the Rad-Vault with the NRF TRIGA casks. Once the mobile crane is positioned and set up, the reception of the transporter carrying two TRIGA casks in impact limiters will be permitted. Attachment of the cask lift beam assembly to the crane will occur after removing the upper impact limiter from one of the TRIGA casks. The TRIGA cask lift beam assembly will be connected to the TRIGA cask and the cask removed from the lower impact limiter. The cask will then be placed in the Rad-Vault. This process will be repeated for the remaining TRIGA casks. Once the Rad-Vault is loaded, the upper impact limiters will be placed on the empty lower impact limiter located on the transporter and the transporter used to carry the TRIGA cask can be released. The lid is then placed on the Rad-Vault.

PlACEMENT OF COMMERCIAL LIGHT WATER REACTOR FUEL CONTAINERS

Operations at the 200 Area ISA start with the receipt and set up of the mobile crane and the receipt and staging of the hydraulic lifting device for the ISO containers. Subsequent to arrival of the transporter carrying a NAC-I cask in an ISO container, the hydraulic lifting device is attached to the mobile crane and then attached to the ISO container. The ISO container is then lifted from the transporter and placed on the assigned position on the storage pad. The transporter can then be released.

Surveillance and Maintenance

There are maintenance activities necessary to ensure long-term storage of the fuel at the ISA (e.g., preventative maintenance including painting of storage cask systems). Minimal, non-safety-related maintenance activities will include periodic inspection of the fence, lighting replacement, and repair, if required. Periodic weed removal is performed to minimize the potential for a fire at the pad.
Interim Storage Cask

Periodic inspection of the ISCs is required to support analysis assumptions. Annual surveillance of the ISC includes a visual inspection, radiation survey, and smear sampling about the ISC environmental cover.

Rad-Vault, NRF TRIGA Cask, and DOT-6M Container

Annual surveillance of the TRIGA cask and DOT-6M containers shall include (1) visual inspection and radiation surveys of the Rad-Vault, (2) vault lid removal to allow visual inspection and radiation surveys of the fuel casks/containers, and (3) smear sampling of the fuel casks/containers and subsequent vault lid placement. As the Rad-Vault is constructed of concrete, an annual visual inspection of the underside is not required.

NAC-1 Casks

Annual surveillance of the NAC-1 casks in ISO containers shall include (1) a visual inspection of the external surface of the ISO container, (2) radiation surveys of the exterior surface of the ISO, (3) a radiation survey of the top of the NAC-1 casks upon opening the ISO container doors, (4) smear samples from the top of the NAC-1 casks, and (5) visual inspections of the interior of the ISO container and exteriors of the NAC-1 casks. As the ISO has structural members constructed of ferrous material, an inspection of the underside will be performed every 5 years to maintain ISO certification. This inspection will be performed in accordance with the requirements of the ISO standards. Operating procedures/work plan will be approved for this activity at that time. Again, critical lift procedures are imposed.

D2.3.3 200 Area Interim Storage Area Confinement

Confinement of radioactive material at the 200 Area ISA is a design feature of each spent fuel storage system. These features are summarized below, with additional details provided in Section D2.6 and Chapter D4.0.

Fast Flux Test Facility Fuel

The ISC is an aboveground concrete and steel shielded, top-loading spent fuel storage cask that is used to provide safe interim dry storage of a CCC (see Figures D2-8 and D2-9) loaded with intact FFTF spent fuel assemblies or pin containers. One CCC can be stored in the cavity of each ISC. The ISC design consists of an all-stainless steel internal confinement structure surrounded by steel and concrete shielding. The fully loaded cask weighs a maximum of 114,200 lbs, including a loaded CCC with a gross payload of 5,000 lbs, the closure hardware, and the weather cover. Outer cask dimensions are 85 in. in diameter and 181 in. tall. The internal cavity of the ISC is 21 in. in diameter and 147 in. tall.
The ISC has been designed and fabricated to meet the requirements of WHC-S-4100, *Specification for FFTF Interim Storage Cask*, in accordance with 10 CFR 72. "Canning" of the spent fuel is provided by the CCC, as discussed in Section D2.6.2.1. The ISC is designed to provide confinement for the fuel, passive heat removal, and environmental protection for the CCC. It also provides radiological shielding protection for site personnel by limiting the dose rate to acceptable levels at normally accessible surfaces. A gasketed weather protection cover is installed on each ISC in the ISA. An additional cover plate may be seal welded over the bolted closure plug after receipt at the 200 Area ISA to enhance the long-term storage configuration.

**NRF TRIGA Fuel**

NRF TRIGA fuel from the NRF in the 300 Area is stored in NRF TRIGA casks and DOT-6M containers. Each NRF TRIGA cask, with capacity to hold up to 18 elements, is nominally 38 in. tall and 16 in. in diameter. Each DOT-6M container, holding one fuel follower control rod (FFCR) element in an inner 2R container, is nominally 70 in. tall and 23 in. in diameter.

Six NRF TRIGA casks and two DOT-6M containers are placed within a concrete Rad-Vault. The Rad-Vault provides environmental protection, supplemental shielding, and natural phenomena hazards resistance. The Rad-Vault is a concrete, vertical, right circular cylinder with light steel reinforcement. The empty Rad-Vault weighs 63,400 lbs., consisting of the 43,400-lb. body and 20,000-lb. lid. The loaded weight of the storage system is approximately 81,400 lbs. With the lid installed, the Rad-Vault is 111 in. tall and the outer diameter is 114 in.

**Commercial Light Water Reactor Fuel**

NAC-I casks within ISO containers are used to store the commercial LWR SNF retrieved from storage at the 324 Building in the Hanford Site 300 Area. Within each NAC-I cask, an inner container provides confinement for the commercial LWR SNF. Each inner container holds either an individual assembly or consolidated pins. The NAC-I cask is a metal cask that provides structural protection and shielding for the canister. Each NAC-I cask is approximated by a right circular cylinder 214 in. long, with a maximum diameter of 50 in. Loaded gross weight of the cask is approximately 49,000 lbs., excluding the weight of the ISO container. The ISO container provides weather protection and has a footprint of 8 ft. by 20 ft.

**D2.3.4 200 Area Interim Storage Area Systems**

This section is not applicable as there are no mechanical or process systems provided by the ISA. Lifting is provided by a portable Manitowoc 4600 250-ton crane.
D2.4 FACILITY STRUCTURE

The 200 Area ISA is designed to DOE Order 6430.1A requirements. It is designated as a nonreactor facility under Section 1300, "Special Facilities," which includes consideration of the design requirements under Section 1320, "Irradiated Fissile Material Storage and Handling Facility." The 200 Area ISA is a temporary facility with a design life of 40 years.

This section contains the structural descriptions for the 200 Area ISA and includes the layout for the various storage pads.

D2.4.1 Interim Storage Area Yard

The 200 Area ISA footprint is nominally 500 ft. by 400 ft. surrounded by a 7-ft. tall chain-link fence, with gates in the fence that control access of vehicles and personnel. The ISA is nominally 710 ft. above sea level. Light poles around the perimeter provide illumination. Within the fenced area are three concrete pads, two for placement of ISCs and one for placement of the NAC-1 casks within ISO containers. The Rad-Vault holding NRF TRIGA casks and DOT-6M containers will be placed on graded, compacted gravel. All concrete pads have embedded conduit that is intended to support any future monitoring needs.

D2.4.2 Equipment Storage Building

An equipment storage building (pre-engineered metal building), approximately 200 ft. to the northeast of the CSB, is proposed near the 200 Area ISA to house lifting devices, impact limiters, transfer casks, and various other types of equipment associated with storage, movement, or transport of the casks. This storage building will be located outside the 200 Area ISA and provides no safety-related function. The proposed storage building consists of a 50 ft. by 85 ft. one-story insulated metal panel wall and roof on a steel frame building. The building will have heating and cooling and a dry pipe fire protection system. Truck and personal access doors will be provided. A fire hydrant is located northeast of the proposed building.

The fence is no closer than 142 ft. to any edge of the NAC-1/ISO concrete pad and 146 ft. to the outer edge of the nearest container storage position on the concrete pad.

Excavation, backfilling, and compacting of building materials are in accordance with American Society for Testing and Materials (ASTM) standards D-653, D-1557, D-2922, and D-3017.

Concrete work on the pads is in accordance with the American Concrete Institute (ACI) standards 117, 224, 301, 306.1, 318, and SP-66; the ASTM A-615, A-853, C-33, C-94, C-150, C-260 and C-881; Hanford HNF-PRO-097, Engineering Design and Evaluation; National Ready
Mix Concrete Association (NRMCA), *Certification of Ready Mixed Concrete Production Facilities*; and the *Uniform Building Code* (ICBO 1994).

Polyvinyl chloride (PVC) externally coated galvanized rigid steel conduit (GRC) meeting the National Electrical Manufacturers Association (NEMA) Standard FW-I, *Polyvinyl Chloride (PVC) Externally Coated Galvanized Rigid Steel Conduit and Intermediate Metal Conduit*, is embedded in the concrete pads. This conduit is listed by the Underwriters Laboratory Electrical Construction Materials directory. The conduits are terminated with a rigid steel coupling and pipe plug, installed flush with the top of the concrete. Conduit ends terminating in vaults or handholds are terminated with ground bushings. Ground bushings are connected in series with a minimum of #12 American Wire Gauge (AWG) bare copper wire to six ground conduits of 0.75 in. diameter.

A fire hydrant is located outside the fence line approximately 152 ft. from the southwest corner of the NAC-1 pad. The hydrant is rated at 1.800 gal/min at 20 lb/in². Drawing H-2-829294 provides the exact location and capacity. There is also a fire hydrant located next to the northeast side of the 200 Area ISA storage building.

**D2.4.3 Fast Flux Test Facility Interim Storage Cask Pad**

The ISC storage pads comprise two pads approximately 171 ft. by 26 ft. by 1.5 ft. thick, in accordance with Drawing H-2-829293, Sheet 4. These pads are reinforced with upper and lower rebar mats (ASTM A-615, Grade 60). The concrete has a minimum compressive strength of 4,000 lbs/in². The pads include embedded electrical conduits in the concrete slabs with handholds (Utility Vault Company Model #3030-LA) installed outside the pad area. The length of these two pads is coincidental with the north-south axis. The two ISC concrete pads are separated by approximately 43 ft. for safety considerations. The western-most edge of the inner ISC concrete pad and the eastern inner edge of the NAC-1/ISO concrete pad are separated by approximately 80 ft.

Each of the ISC pads are intended to support storage of 30 ISC containers. The storage locations are two-abreast. Each position has a nominal circular contact footprint 7 ft. in diameter. There are 11 ft. between the center lines of each adjacent container. There is a minimum of 4 ft. between the edges of each storage position, and 4 ft. between each outer edge and the edge of the ISC concrete pad. A representation of the 200 Area ISA pad locations relative to the CSB are provided in Figure D2-1.

**D2.4.4 NRF TRIGA Rad-Vault Gravel Area**

The Rad-Vault will not be placed on a concrete pad; instead, a graded and compacted gravel area will be prepared by the project. The design specifications for this gravel area are
provided in Drawing H-2-829293. The gravel will be compacted to a soil-bearing capacity greater than 4,000 lbs/ft².

**D2.4.5 Commercial Light Water Reactor International Standards Organization Container Pad**

The third storage pad is approximately 88 ft. by 28 ft. by 1 ft. thick, in accordance with Drawing H-2-829293, Sheet 3 Rev. as-built. This pad is reinforced with upper and lower rebar mats (ASTM A-615, Grade 60). The concrete has a minimum compressive strength of 4,000 lbs/in². The pad includes embedded electrical conduits in the concrete slab with handholds (Utility Vault Company Model #3030-LA) installed outside the pad area. The length of this pad is coincidental with the north-south axis.

The NAC-1/ISO container pad is designed to support storage of seven containers. The storage locations are sequential with 4 ft. of space between each storage ISO container edge and 4 ft. to the edge of the concrete pad. Each position has a nominal footprint of 8 ft. x 20 ft.

**D2.4.6 Design Basis**

With more than one type of dry cask storage system stored within the ISA, specific criteria that each dry cask storage system must address will be provided as separate subsections. This section delineates the generic requirements, limits, and siting criteria that must be met by all fuel storage systems residing within the ISA independent of the fuel type and specific design.

A summary of each ISA facility design requirement and its justification or reference is presented in Table D2-2. As fuel storage systems are added, their subsection will address the outline of requirements and limits provided in this section. This will ensure that consistency of the SNF Project FSAR authorization basis is maintained and will demonstrate each cask system’s acceptability for safe, dry spent fuel storage at the 200 Area ISA.

Because the ISA provides only the general-service functions described in Section D2.3, the cask systems are the major components relied on for meeting the 10 CFR 72 requirements for safe storage of spent fuel. Accordingly, each cask storage configuration provides all the necessary confinement, shielding, criticality control, passive heat removal characteristics, and natural phenomena hazards resistance necessary for the specific spent fuel configuration to be stored.
Table D2-2. Principal Design Requirements for 200 Area Interim Storage Area. (4 sheets)

<table>
<thead>
<tr>
<th>Category</th>
<th>Requirement</th>
<th>Justification/Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Radiological Dose Limits</td>
<td>0.05 mrem/hr at the fence</td>
<td>HSRCM-1 limit for uncontrolled access by the public.</td>
</tr>
<tr>
<td></td>
<td>60 mrem/hr on contact</td>
<td>HSRCM-1 derived limit for 8 hours exposure per year.</td>
</tr>
<tr>
<td>Accident Limits</td>
<td>On-site</td>
<td>Sellers 1995 (95-SFD-167)</td>
</tr>
<tr>
<td></td>
<td>1 rem (10^4 &gt; frequency &gt; 10^2)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>10 rem (10^3 &gt; frequency &gt; 10^4)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>25 rem (10^4 &gt; frequency &gt; 10^6)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Off-site</td>
<td></td>
</tr>
<tr>
<td></td>
<td>500 mrem (10^1 &gt; frequency &gt; 10^2)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>5 rem (10^3 &gt; frequency &gt; 10^4)</td>
<td></td>
</tr>
<tr>
<td></td>
<td>5 rem (10^4 &gt; frequency &gt; 10^6)</td>
<td></td>
</tr>
<tr>
<td>Seismic</td>
<td>0.26 g ZPA horizontal, 2/3 of horizontal acceleration in the vertical direction</td>
<td>HNF-SD-SNF-DB-009, Appendix C</td>
</tr>
<tr>
<td>Wind</td>
<td>80 mph fastest mile wind</td>
<td>HNF-SD-SNF-DB-009, Table 1</td>
</tr>
<tr>
<td></td>
<td>Steady state - 70 mph</td>
<td></td>
</tr>
<tr>
<td>Wind Missile</td>
<td>15-lb 2x4 missile at 50 mph</td>
<td>HNF-SD-SNF-DB-009, Section 3.2</td>
</tr>
<tr>
<td>Tornado</td>
<td>200 mph resultant wind speed</td>
<td>HNF-SD-SNF-DB-009, Table 1</td>
</tr>
<tr>
<td></td>
<td>Differential pressure - 0.90 lbs/in^2 over 3 seconds</td>
<td>HNF-SD-SNF-DB-009, Section 3.3</td>
</tr>
<tr>
<td>Tornado Missile</td>
<td>NA (probability &lt; 10^-6)</td>
<td>HNF-1785</td>
</tr>
<tr>
<td>Flood</td>
<td>Not applicable</td>
<td>Analysis shows the 200 Area is above the flood-affected zone (HNF-SD-SNF-DB-009)</td>
</tr>
<tr>
<td>Rain</td>
<td>Site drainage: 9.2 in. for 6-hour PMP</td>
<td>HNF-SD-SNF-DB-009, Table 1</td>
</tr>
</tbody>
</table>
### Table D2-2. Principal Design Requirements for 200 Area Interim Storage Area. (4 sheets)

<table>
<thead>
<tr>
<th>Category</th>
<th>Requirement</th>
<th>Justification/Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ash</td>
<td>24 lb/ft²</td>
<td>HNF-SD-SNF-DB-009 imposes an additional roof load for ashfall. Ashfall shall also be addressed as an insulator during heat load calculations.</td>
</tr>
<tr>
<td>Snow</td>
<td>20 lb/ft²</td>
<td>HNF-SD-SNF-DB-009, Table 1</td>
</tr>
<tr>
<td>Soil Loads</td>
<td>Soil density of 110 lb/ft² is assumed.</td>
<td>Drawing H-2-829293, &quot;A Civil Site Plan for ISA&quot;</td>
</tr>
<tr>
<td></td>
<td>Soil bearing capacity = 2,500 lb/ft²</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Concrete design of foundations:</td>
<td></td>
</tr>
<tr>
<td></td>
<td>UBC and ACI-318 (include dynamic earth pressures)</td>
<td></td>
</tr>
<tr>
<td>Lightning</td>
<td>Lightning protection shall be provided for the ISA to meet NFPA 780 (if the container cannot withstand a direct strike).</td>
<td>HNF-SD-SNF-DB-009, Table 1</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Lightning protection required by 10 CFR 72, if container cannot withstand a strike.</td>
</tr>
<tr>
<td>Fire</td>
<td>Storage systems shall be designed to withstand the 10 CFR 71.73(c)(3) transportation fire. Defined as a 1,475 °F fire fully engulfed for 30 minutes with return to ambient temperature without water quenching/cooling.</td>
<td>200 Area ISA FHA (SNF-4932) considers the fire analysis prepared for 10 CFR 71.73(c)(3) to be bounding for all scenarios. 10CFR72 also defaults to this fire.</td>
</tr>
<tr>
<td>Load Combinations</td>
<td>Load combinations and allowable stresses for live, dead, snow and normal operating loads shall be applied.</td>
<td>UBC</td>
</tr>
<tr>
<td></td>
<td>Load combinations and allowable stresses for normal operating loads and natural phenomenon loads.</td>
<td>UCRL-15910 (replaced by DOE-STD-1020-94)</td>
</tr>
<tr>
<td></td>
<td>Load combinations for original cask construction.</td>
<td>NRC Regulatory Guide 7.6, ASME Code, Section III</td>
</tr>
</tbody>
</table>
### Table D2-2. Principal Design Requirements for 200 Area Interim Storage Area. (4 sheets)

<table>
<thead>
<tr>
<th>Category</th>
<th>Requirement</th>
<th>Justification/Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal</td>
<td>Decay heat removal shall be by passive means only.</td>
<td>No operating safety systems are planned for the ISA due to the passive heat removal cask design.</td>
</tr>
<tr>
<td></td>
<td>Maximum surface temperature under most severe environmental conditions shall not exceed 185 °F.</td>
<td>Temperature limit per 10 CFR 71 (revised April 1996).</td>
</tr>
<tr>
<td></td>
<td>Fuel temperature shall not exceed design basis temperatures.</td>
<td>General requirement for all fuel storage systems.</td>
</tr>
<tr>
<td></td>
<td>Cask components with specified temperature limits (i.e., seals) shall not exceed design basis temperatures.</td>
<td>General requirement for all fuel storage systems.</td>
</tr>
<tr>
<td></td>
<td>Environmental temperature range from -27 °F to +115 °F.</td>
<td>Canister Storage Building design basis document.</td>
</tr>
<tr>
<td>Internal Pressure</td>
<td>The storage system shall be designed to withstand the worst credible internal pressure generation scenario identified during accident analysis. Testing shall be to 125 - 150% of design per the ASME Code.</td>
<td>General requirement to ensure that outdoor environmental conditions, including direct sunlight, are included in design and accident scenarios involving gas pressure events along with 100% rod rupture.</td>
</tr>
<tr>
<td>Radiation</td>
<td>Isotopic composition of the stored material shall be defined.</td>
<td>Composition shall be determined either by analysis or calculation.</td>
</tr>
<tr>
<td>Source</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Shielding</td>
<td>Storage configuration shall meet the requirement of 60 mrem/hr external surface contact dose rate in the storage configuration.</td>
<td>HSRCM-1 limit of 500 mrem/yr and ALARA considerations for 8 hr/yr exposure and 10 CFR 835 requirements.</td>
</tr>
<tr>
<td>Structural</td>
<td>Systems shall be designed to withstand the conditions as required for their designated safety classification.</td>
<td>The safety classification system requires that SSCs survive specified natural phenomenon events. More severe accident events must not result in consequences that exceed the guidelines identified for normal and abnormal events.</td>
</tr>
</tbody>
</table>
Table D2-2. Principal Design Requirements for 200 Area Interim Storage Area. (4 sheets)

<table>
<thead>
<tr>
<th>Category</th>
<th>Requirement</th>
<th>Justification/Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Handling Loads</td>
<td>The storage system and/or components shall be designed to be lifted by crane.</td>
<td>General requirement for the ISA.</td>
</tr>
<tr>
<td></td>
<td>Systems and components shall be designed to withstand a set down load on rigid surfaces at impact velocities of up to 10 ft/min. (~2 g).</td>
<td>This simulates a hard set down by the crane.</td>
</tr>
<tr>
<td></td>
<td>Systems and components shall be designed to withstand the maximum handling drop anticipated.</td>
<td>Systems and components must be analyzed to withstand the maximum handling drop anticipated.</td>
</tr>
<tr>
<td>Criticality</td>
<td>$k_{eff} \leq 0.95$</td>
<td>NRC equivalence.</td>
</tr>
</tbody>
</table>

ALARA = as low as reasonably achievable.
ASME = American Society of Mechanical Engineers.
FHA = fire hazards analysis.
HSRCM = Hanford Site Radiological Control Manual.
ISA = Interim Storage Area.
NA = not applicable.
NRC = U.S. Nuclear Regulatory Commission.
SSC = structure, system, and component.
UBC = Uniform Building Code.
ZPA = zero-period acceleration.
D2.5 PROCESS DESCRIPTION

The 200 Area ISA has no active processing systems. The information provided in this section includes fuel descriptions and parameters.

D2.5.1 Fuel Description

Table D2-3 summarizes the fuel history and fuel specifics for each SNF dry storage system.

D2.5.1.1 Fast Flux Test Facility Fuel. The purpose of the FFTF was to provide testing capability to satisfy the diverse technology development needs for the advanced reactor programs. The mission included irradiation and evaluation of different types of fuel assemblies and different materials for fuel assembly construction. Also included was direct production of useful materials such as medical isotopes. In December 1993, DOE directed that the FFTF be transitioned to shutdown status. In April 1994, removal of fuel from the reactor began.

There are nominally 374 fueled components at FFTF that will be placed into storage at the 200 Area ISA. Since cessation of FFTF reactor operations in April 1992, the fuel has decayed and a substantial reduction in associated fission products and noble gases has occurred. As of August 1995, 351 fueled components were below 200 W decay heat; the highest decay heat assembly was 329 W. The average decay heat, considering all fuel components, is 81 W. All assemblies currently are below 250 W decay heat.

All fuel assemblies fall into two general categories; driver fuel assemblies (DFAs) and test assemblies. The primary purpose of the DFAs was to hold the fuel that creates the neutron flux environment to carry out the various FFTF tests. The primary purpose of the test assemblies was to hold the fuel and non-fuel materials being tested. The fueled test assemblies also contributed to the neutron flux within the reactor.

The test assemblies have been further categorized for the purpose of this document. The test assemblies that are identical to or very similar to the DFAs are categorized as test DFAs. The test assemblies that contain experimental fuel are categorized as test fuel assemblies, even if they also contain some standard driver fuel pins. The remaining test assemblies did not contain fuel.
Table D2-3. Summary of Fuel Characteristics.

<table>
<thead>
<tr>
<th>Fuel type</th>
<th>Fast Flux Test Facility</th>
<th>TRIGA cask</th>
<th>TRIGA DOT-6M</th>
<th>Commercial Light Water Reactor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel type</td>
<td>FFTF driver fuel and experimental assemblies</td>
<td>TRIGA fuel element</td>
<td>FFCR</td>
<td>PWR and BWR</td>
</tr>
<tr>
<td>Maximum enrichment</td>
<td>29.3% Pu(^{(1)})</td>
<td>ZrII with 8.0 - 8.5 wt% U - 20% (^{235})U Enriched</td>
<td>&lt;1%</td>
<td>PWR 3.6%/BWR 3.06%</td>
</tr>
<tr>
<td>Burnup</td>
<td>150,000 MWd/MTHM(^{(2)})</td>
<td>&lt;1%</td>
<td>&lt;1%</td>
<td>16.92 kg (^{235})U max</td>
</tr>
<tr>
<td>Maximum heat</td>
<td>1500 watts</td>
<td>2.7 watts total in the Rad-Vault</td>
<td></td>
<td>12.4 kW</td>
</tr>
<tr>
<td>Maximum fuel load per dry storage system</td>
<td>7 DFA, or 6 Ident-69, or 5 Ident-69 and 2 DFAs</td>
<td>18 fuel elements</td>
<td>1 FFCR</td>
<td>1 PWR assembly or 1 consolidated pin container</td>
</tr>
<tr>
<td>Condition of fuel</td>
<td>Intact</td>
<td>Intact</td>
<td>Intact</td>
<td>Intact</td>
</tr>
</tbody>
</table>

\(^{(1)}\) Enrichments greater than 29.3% must be analyzed on a case-by-case basis.

\(^{(2)}\) Burnups greater than 150,000 MWd/MTHM are analyzed on a case-by-case basis.

BWR = boiling water reactor.
DFA = driver fuel assemblies.
DOT = U.S. Department of Transportation.
FFCR = fuel follower control rod.
FFTF = Fast Flux Test Facility.
MTHM = metric ton of heavy metal.
PWR = pressurized water reactor.
TRIGA = Training, Research and Isotope Production, General Atomics.
The following FFTF fuel types have been authorized for storage at the 200 Area ISA (based on fuel previously authorized for storage at the FFTF 400 Area ISA):

- Standard DFAs.
- Driver evaluation, core characterizer assemblies, and run-to-cladding-breach assemblies are also included based on their similarities to a standard DFA.
- Experimental assemblies and Ident-69 pin containers listed in Table D2-4 have been authorized based on the analyses summarized in WHC-SD-FF-RPT-005, *Review of FFTF Fuel Experiments for Storage at ISA*. Table D2-5 shows experimental assemblies not listed in Table D2-4 that have been disassembled and loaded into Ident-69s.

Table D2-4. Experimental Fuel Assemblies and Ident-69 Pin Containers Analyzed for Loading into a Core Component Container/Interim Storage Cask.

<table>
<thead>
<tr>
<th>Experimental assemblies</th>
<th>Ident-69 pin containers</th>
</tr>
</thead>
<tbody>
<tr>
<td>AAD-1, 2, 3, 4, 6, 7</td>
<td>ID-69 S/N 1</td>
</tr>
<tr>
<td>ACO-2, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16</td>
<td>ID-69 S/N 2</td>
</tr>
<tr>
<td>AW-1</td>
<td>ID-69 S/N 3</td>
</tr>
<tr>
<td>CRBR-1, 3, 5</td>
<td>ID-69 S/N 5</td>
</tr>
<tr>
<td>CV-1</td>
<td>ID-69 S/N 7</td>
</tr>
<tr>
<td>D9-3</td>
<td>ID-69 S/N 8</td>
</tr>
<tr>
<td>DE-HTD</td>
<td>ID-69 S/N 9</td>
</tr>
<tr>
<td>DEA-2</td>
<td>ID-69 S/N 10</td>
</tr>
<tr>
<td>DIPRESS</td>
<td>ID-69 S/N 11</td>
</tr>
<tr>
<td>FOTA-1, 2</td>
<td>ID-69 S/N 12</td>
</tr>
<tr>
<td>GF001, GF002</td>
<td>ID-69 S/N 13</td>
</tr>
<tr>
<td>MW-1, 2, 3, 4, 5, 6</td>
<td>ID-69 S/N 14</td>
</tr>
<tr>
<td>PO-2, 5</td>
<td>ID-69 S/N 15</td>
</tr>
<tr>
<td>RNTT-1</td>
<td>ID-69 S/N 17</td>
</tr>
<tr>
<td>RTCB-4</td>
<td>ID-69 S/N 18</td>
</tr>
<tr>
<td>SRF-1, 2</td>
<td>ID-69 S/N 21</td>
</tr>
<tr>
<td>WF004, WF005</td>
<td>ID-69 S/N 22</td>
</tr>
<tr>
<td>Ident-69 S/N</td>
<td>Type(1)</td>
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<tr>
<td>-------------</td>
<td>---------</td>
</tr>
<tr>
<td>1</td>
<td>69C</td>
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<td>27</td>
<td>69C</td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td>29</td>
<td>69C</td>
</tr>
<tr>
<td>30</td>
<td>69C</td>
</tr>
<tr>
<td>31</td>
<td>69C</td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Table D2-5. Pin Container Test Pin Contents. (2 sheets)

<table>
<thead>
<tr>
<th>Ident-69 S/N</th>
<th>Type(^{(1)})</th>
<th>Total pins</th>
<th>Original component</th>
<th>Number of pins</th>
<th>Burnup (MWd/MTHM)</th>
<th>Decay heat (W) 3/06/96</th>
</tr>
</thead>
<tbody>
<tr>
<td>32</td>
<td>69C</td>
<td>146</td>
<td>4123 RTCB-7</td>
<td>15</td>
<td>70,840</td>
<td>87.12</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>16497 C-1(^{(1)})</td>
<td>79</td>
<td>88,050</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>16523 D9-4(^{(2)})</td>
<td>52</td>
<td>102,360</td>
<td></td>
</tr>
<tr>
<td>33</td>
<td>69C</td>
<td>151</td>
<td>2171 MFA-2(^{(2)})</td>
<td>151</td>
<td>127,780</td>
<td>183.26</td>
</tr>
<tr>
<td>35</td>
<td>69C</td>
<td>135</td>
<td>16523 D9-4(^{(2)})</td>
<td>135</td>
<td>102,360</td>
<td>88.59</td>
</tr>
<tr>
<td>37</td>
<td>69C</td>
<td>128</td>
<td>8252 ACO-1(^{(3)})</td>
<td>128</td>
<td>93,920</td>
<td>89.59</td>
</tr>
<tr>
<td>38</td>
<td>69C</td>
<td>126</td>
<td>16489 AAD-5(^{(2)})</td>
<td>57</td>
<td>73,860</td>
<td>83.58</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>16505 D9-2(^{(2)})</td>
<td>69</td>
<td>127,360</td>
<td></td>
</tr>
<tr>
<td>39</td>
<td>69C</td>
<td>134</td>
<td>2065 PO-1(^{(2)})</td>
<td>49</td>
<td>88,310</td>
<td>82.22</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>2155 MFF-LA(^{(28,5)})</td>
<td>85</td>
<td>46,060</td>
<td></td>
</tr>
<tr>
<td>40</td>
<td>69C</td>
<td>126</td>
<td>2144 ACO-3(^{(2)})</td>
<td>126</td>
<td>197,450</td>
<td>208.19</td>
</tr>
<tr>
<td>42</td>
<td>69</td>
<td>108</td>
<td>16426 DE-1-2</td>
<td>66</td>
<td>21,360</td>
<td>44.69</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>16454 DEA-1</td>
<td>42</td>
<td>2,860</td>
<td></td>
</tr>
<tr>
<td>43</td>
<td>69</td>
<td>109</td>
<td>16454 DEA-1</td>
<td>109</td>
<td>2,860</td>
<td>40.59</td>
</tr>
<tr>
<td>44</td>
<td>69C</td>
<td>139</td>
<td>2143 FO-2(^{(2)})</td>
<td>16</td>
<td>46,840</td>
<td>95.11</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>8252 ACO-1(^{(2)})</td>
<td>16</td>
<td>93,920</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>16505 D9-2(^{(2)})</td>
<td>107</td>
<td>127,360</td>
<td></td>
</tr>
<tr>
<td>45</td>
<td>69C</td>
<td>113</td>
<td>2143 FO-2(^{(2)})</td>
<td>113</td>
<td>46,840</td>
<td>64.16</td>
</tr>
<tr>
<td>46</td>
<td>69C</td>
<td>72</td>
<td>2071 AC-3(^{(2)})</td>
<td>72</td>
<td>66,690</td>
<td>101.99</td>
</tr>
<tr>
<td>x</td>
<td>x</td>
<td>x</td>
<td>2069 Po-4(^{(3)})</td>
<td>x</td>
<td>80,000</td>
<td>126.00</td>
</tr>
<tr>
<td>x</td>
<td>x</td>
<td>x</td>
<td>2111 ACN - 1(^{(3)})</td>
<td>x</td>
<td>67,450</td>
<td>89.99</td>
</tr>
</tbody>
</table>

\(^{(1)}\) Maximum pins per type: 69 = 109, 69A = 55, 69C = 217.

\(^{(2)}\) Experimental fuel pins.

\(^{(3)}\) PO-4 and ACN-1 are currently intact assemblies but will be disassembled and placed in pin containers.

\(^{(4)}\) IFR-1 pins are sodium-bonded metal fuel.

\(^{(5)}\) The MFF-LA pins stored at the Fast Flux Test Facility are mixed oxide; all of the sodium-bonded pins from this assembly were shipped to Argonne National Laboratory - West.
The remaining fuel and test assemblies require further evaluation before they can be accepted at the ISA. An Engineering Change Notice to this document is required to authorize the additional fuel assemblies listed in Table D2-6.

### Table D2-6. Assemblies That Require Further Evaluation.

<table>
<thead>
<tr>
<th>Fuel Category</th>
<th>Test Assembly Name</th>
<th>Evaluations Required</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sodium bonded metal fuel</td>
<td>MFF-1, MFF-2, MFF-3, MFF-4, MFF-5, MFF-6, IFR-1 (I.D. 69 #4), MFF-8A</td>
<td>Radiological accident release</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Shielding</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Criticality (MFF-2 through MFF-6 and MFF-8A only)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Design fuel pin temperature limit</td>
</tr>
<tr>
<td>Carbide fuel</td>
<td>FC-1, AC-3, ACN-1</td>
<td>Radiological accident release (ACN-1 only)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Criticality (ACN-1 only)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Thermal (ACN-1 if the assembly remains intact or partially intact [individual pins are acceptable])</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Shielding</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Design fuel pin temperature limit</td>
</tr>
<tr>
<td>Delayed neutron monitor signal/encapsulated failed pins</td>
<td>ACN-1, PO-4, PB19#6C (from DFA-16392)</td>
<td>Evaluation of cleaning method and acceptability for storage</td>
</tr>
<tr>
<td>Highly enriched fuel</td>
<td>SRF-3, SRF-4(I)</td>
<td>Criticality</td>
</tr>
<tr>
<td>Blanket assemblies</td>
<td>ABA-1, ABA-2, ABA-3, ABA-4, ABA-5, ABA-6, WBA-40, WBA-41, MBA-1 (I.D. 69 #29 and pin basket PB19C), UO-1, AB-1</td>
<td>Design fuel pin temperature limit</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Shielding</td>
</tr>
</tbody>
</table>

(1) SRF-3 and SRF-4 have a thermal limit of 62.5 watts for storage in the core component container/interim storage tank.
As noted in Section D6.3.3.1.1, various FFTF experimental pins, cropped fuel pins, and fuel drillings are to be returned to FFTF. Fuel debris is not currently authorized for storage at the 200 Area ISA, as FFTF fuel received at the ISA is required to have intact cladding upon receipt. Experimental fuel debris has been shown acceptable with respect to criticality safety in Chapter D6.0 even though it is not currently authorized for storage at the ISA. However, impacts of fuel debris on radiological accident releases must still be evaluated.

There are 210 DFAs, 65 test DFAs, and 54 test fuel assemblies irradiated at the FFTF. Of these, 1 DFA, 14 test DFAs, and 13 test fuel assemblies have been disassembled. Also, there were 55 DFAs, 1 test DFA, and 2 test fuel assemblies that had been fabricated but not irradiated. Unirradiated FFTF fuel is not permitted to be stored at the 200 Area ISA for safeguards consideration.

There are 8.6 metric tons of uranium (MTU) and 2.4 metric tons of plutonium in the FFTF fuel, as shown in Table D2-7. The plutonium and uranium content of the individual assemblies varies depending on the assembly exposure and its pre-irradiation content.

<table>
<thead>
<tr>
<th>Fuel Type</th>
<th>Uranium (MT)</th>
<th>Plutonium (MT)</th>
<th>Total (MT)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Driver Fuel Assemblies</td>
<td>5.2</td>
<td>1.6</td>
<td>6.8</td>
</tr>
<tr>
<td>Test Driver Fuel Assemblies</td>
<td>1.7</td>
<td>0.5</td>
<td>2.2</td>
</tr>
<tr>
<td>Test Fuel Assemblies</td>
<td>1.7</td>
<td>0.3</td>
<td>2.0</td>
</tr>
<tr>
<td>Total</td>
<td>8.6</td>
<td>2.4</td>
<td>11.0</td>
</tr>
</tbody>
</table>

The driver fuel is plutonium-uranium mixed oxide enriched in fissile plutonium (239 and 241). The essential design features that affect performance are as follows:

- Fuel column length: 36 in.
- Cladding outside diameter: 0.230 in.
- Cladding thickness: 0.015 in.
- Beginning of life (BOL) diametrical gap/pellet diameter: 0.0055/0.1945 in.
- Smear density: 85.5% theoretical density
- Fill gas composition and pressure: helium (plus tag gas), 1 atmosphere at room temperature
The fuel pins contain mixed uranium/plutonium oxide clad in 20% cold-worked Type 316 stainless steel and are wire wrapped to maintain pin spacing. The 217 pins are oriented on a triangular pitch in the subassembly with a spacer wire 0.056 in. in diameter.

**Driver Fuel Assembly**

The DFAs are hexagonally shaped components, which are composed of 217 fuel pins, a surrounding duct, a shield orifice assembly, an inlet nozzle, load pads, and a handling socket. The assembly is 12 ft. long, 4.575 in. wide across the hexagon flats, 5.16 in. wide across the hexagon points, and weighs 38.1 lbs.

The principle structural component is the Type 316 stainless steel, hexagonal duct that extends from a handling socket at the top, to a shield/orifice region located below the fuel pins. The duct wall thickness is nominally 0.120 in., but varies from 0.065 in. through 0.190 in. along the length to facilitate lateral positioning of assemblies within the core and the distribution of force among the assemblies within the core.

The fuel pins are 0.23 in. in diameter, approximately 93.5 in. long, and have a 36-in. fuel bearing region that is centered 65.5 in. from the bottom end of the fuel assembly. Each fuel pin is helically wrapped with a 0.056-in. diameter steel wire to provide lateral spacing along its length. The fuel region contains approximately 150 pressed and sintered, mixed uranium-plutonium oxide pellets. Enrichment details are shown in Table D2-3.

**Test Driver Fuel Assemblies**

The test DFAs are either identical or very similar to the standard DFA. Each contains fuel with only one of the standard four plutonium enrichments used in the DFA. Some test DFAs have components made of D9 or HT9 stainless steels versus Type 316 stainless steel.

Fuel pins were removed from some test DFAs and shipped off-site for examination. Some of these removed fuel pins were returned whole in Ident-69 containers or as pieces in a cask. Specific details are provided in WHC-SD-SNF-TI-001, *Hanford Spent Fuel Inventory Baseline*. Only intact pins will be stored in Ident-69 containers.

**D2.5.1.2 NRF TRIGA Fuel.** A 250 kW TRIGA experimental research reactor was operated in the 300 Area intermittently from the late 1970s until its last power run in May 1989. The reactor was manufactured by Gulf General Atomics Company, Incorporated, of San Diego, California, and was used primarily for neutron radiography of FFTF fuel elements and test assemblies. The fuel from the reactor core/pool storage has been removed as part of the decommissioning of the facility. The NRF TRIGA irradiated fuel inventory consists of 99 TRIGA fuel elements and two FFCRs. The standard fuel elements are stored in the NRF TRIGA casks, and the FFCRs are stored in DOT-6M containers. The NRF TRIGA casks and the DOT-6M containers are stored in a Rad-Vault, as described in Section D2.3.2. The fuel elements are clad with either aluminum or Type 304 stainless steel. The FFCRs are clad with Type 304 stainless steel.
There are a total of 66 aluminum clad elements. Fifty-six elements were used in the NRF TRIGA core for various periods of time during the thirteen years of operation, one was rejected in the first year because of dimensional changes (bowing), and nine were never used in the NRF TRIGA core. All of these aluminum clad fuel elements were received previously irradiated. There were 33 fuel elements received from the Armed Forces Fuel Research Institute Reactor and 33 from the Nevada Test Site.

There are a total of 33 stainless steel clad fuel elements. Twelve elements were new from General Atomics and were used in the core for the entire thirteen years. Twenty elements were received from the deactivated Diamond Ordnance Reactor Facility; however, none of these elements were used in the 300 Area reactor core. One element was an instrumented fuel element from Oregon State University, also never used in the 300 Area reactor core. This instrumented fuel element had a thermocouple tube attached at the upper end. This 8-in. long tubing was bent to 90° with a 0.25 in. radius so it would not interfere with the closure lid of the inner container. There are two stainless steel clad FFCRs that also were previously irradiated at the Diamond Ordnance Reactor Facility reactor but never used in the 300 Area reactor core.

The design parameters of the TRIGA fuel are summarized in Table D2-8. All the fuel elements are 28.37 in. long and 1.47 in. in diameter. The aluminum clad elements have a fuel core length of 14 in., while the stainless steel clad elements have a core length of 15 in. The FFCRs are 45.75 in. long and 1.375 in. in diameter. The fuel length for the FFCR is 15 in. The FFCR also includes a borated graphite column above the fuel core that is 14.24 in. long.

The fuel columns in all fuel assemblies are ceramic zirconium hydride with 8 to 8.5 wt% uranium that is nominally 20 percent enriched $^{235}U$ dispersed throughout the hydride. The $^{235}U$ content of the fuel assemblies before irradiation ranged from 32 to 40.5 g each. The FFCR typically contained 32 g $^{235}U$ each. Table D2-9 lists each fuel element and FFCR, including serial number, operating hours in the 300 Area TRIGA reactor, kWh (exposure), fuel source, date received, cladding type, fuel column length, and gram quantity of $^{235}U$. The estimated dose rates, based on reported comparisons with underwater measurements in the NRF TRIGA pool, are for single TRIGA fuel elements at midplane. The irradiated fuel elements have limited burn-up from use in the 300 Area TRIGA reactor; the $^{235}U$ burn-up is probably less than one percent (30 to 35 g) of the total grams of $^{235}U$ in all of the fuel elements.
Annex D - 200 Area Interim Storage Area

Table D2-8. Principal Design Parameters for NRF TRIGA Fuel.

<table>
<thead>
<tr>
<th>Design parameter</th>
<th>Stainless steel clad</th>
<th>Aluminum clad</th>
<th>Fuel follower control rod</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel moderated material</td>
<td>UZrH₁₆</td>
<td>UZrH₁₆</td>
<td>UZrH₁₆</td>
</tr>
<tr>
<td>Uranium content</td>
<td>8.5 wt%</td>
<td>8.0 wt%</td>
<td>8.5 wt%</td>
</tr>
<tr>
<td>Uranium enrichment</td>
<td>~20% ²³⁵U</td>
<td>~20% ²³⁵U</td>
<td>~20% ²³⁵U</td>
</tr>
<tr>
<td>Shape of fuel core</td>
<td>Cylindrical</td>
<td>Cylindrical</td>
<td>Cylindrical</td>
</tr>
<tr>
<td>Length of fuel core</td>
<td>15 in.</td>
<td>14 in.</td>
<td>15 in.</td>
</tr>
<tr>
<td>Diameter of fuel core (OD)</td>
<td>1.43 in.</td>
<td>1.41 in.</td>
<td>1.335 in.</td>
</tr>
<tr>
<td>²³⁵U content (Nominal)</td>
<td>38 g</td>
<td>37 g</td>
<td>32 g</td>
</tr>
<tr>
<td>Fuel mixture (atomic ratio)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Zr</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>H</td>
<td>1.6</td>
<td>1.0</td>
<td>1.6</td>
</tr>
<tr>
<td>U (wt%)</td>
<td>8.5</td>
<td>8.0</td>
<td>8.5</td>
</tr>
<tr>
<td>Zirconium hydride content</td>
<td>91.5 wt%</td>
<td>92.0 wt%</td>
<td>91.5 wt%</td>
</tr>
<tr>
<td>Length of top and bottom graphite reflectors</td>
<td>3.47 in.</td>
<td>4.00 in.</td>
<td>14.25 in. (length of borated graphite column)</td>
</tr>
<tr>
<td>Cladding material</td>
<td>Type 304 SS</td>
<td>Al-1100, anodized</td>
<td>Type 304 SS</td>
</tr>
<tr>
<td>Top and bottom end fixture material</td>
<td>Type 304 SS</td>
<td>Al-1100, anodized</td>
<td>Type 304 SS</td>
</tr>
<tr>
<td>Cladding thickness</td>
<td>0.020 in.</td>
<td>0.030 in.</td>
<td>0.020 in.</td>
</tr>
<tr>
<td>Diameter (OD)</td>
<td>1.47 in.</td>
<td>1.47 in.</td>
<td>1.375 in.</td>
</tr>
<tr>
<td>Overall length</td>
<td>28.37 in.</td>
<td>28.37 in.</td>
<td>45.75 in.</td>
</tr>
<tr>
<td>Number of elements</td>
<td>33</td>
<td>66</td>
<td>2</td>
</tr>
</tbody>
</table>

NRF = Neutron Radiography Facility.
OD = outer dimension.
SS = stainless steel.
TRIGA = Training, Research and Isotope Production, General Atomics.
<table>
<thead>
<tr>
<th>Fuel element serial</th>
<th>Operation core (kilowatt hours)</th>
<th>NRF Core operating hours</th>
<th>Received from</th>
<th>Date received</th>
<th>Fuel element cladding</th>
<th>Fuel column length (in.)</th>
<th>Original $^{235}$U content (g)</th>
</tr>
</thead>
<tbody>
<tr>
<td>2463</td>
<td>0.000</td>
<td>0.000</td>
<td>AFFRI</td>
<td>7/74</td>
<td>AL</td>
<td>14</td>
<td>39.93</td>
</tr>
<tr>
<td>2474</td>
<td>0.000</td>
<td>0.000</td>
<td>AFFRI</td>
<td>8/74</td>
<td>AL</td>
<td>14</td>
<td>38.44</td>
</tr>
<tr>
<td>2477</td>
<td>0.000</td>
<td>0.000</td>
<td>AFFRI</td>
<td>7/74</td>
<td>AL</td>
<td>14</td>
<td>39.81</td>
</tr>
<tr>
<td>2558</td>
<td>0.000</td>
<td>0.000</td>
<td>AFFRI</td>
<td>7/74</td>
<td>AL</td>
<td>14</td>
<td>39.22</td>
</tr>
<tr>
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<td>0.000</td>
<td>0.000</td>
<td>AFFRI</td>
<td>7/74</td>
<td>AL</td>
<td>14</td>
<td>38.87</td>
</tr>
<tr>
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<td>0.000</td>
<td>0.000</td>
<td>AFFRI</td>
<td>7/74</td>
<td>AL</td>
<td>14</td>
<td>39.12</td>
</tr>
<tr>
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<td>0.000</td>
<td>0.000</td>
<td>AFFRI</td>
<td>7/74</td>
<td>AL</td>
<td>14</td>
<td>40.34</td>
</tr>
<tr>
<td>2336</td>
<td>185,434.960</td>
<td>806.760</td>
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<td>8/74</td>
<td>AL</td>
<td>14</td>
<td>39.38</td>
</tr>
<tr>
<td>3077</td>
<td>211,065.867</td>
<td>1,106,259</td>
<td>AFFRI</td>
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<td>AL</td>
<td>14</td>
<td>36.82</td>
</tr>
<tr>
<td>2610</td>
<td>316,012.072</td>
<td>1,648.169</td>
<td>AFFRI</td>
<td>7/74</td>
<td>AL</td>
<td>14</td>
<td>39.69</td>
</tr>
<tr>
<td>3073</td>
<td>316,012.072</td>
<td>1,648.169</td>
<td>AFFRI</td>
<td>7/74</td>
<td>AL</td>
<td>14</td>
<td>35.83</td>
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<tr>
<td>2335</td>
<td>352,915.235</td>
<td>1,662.330</td>
<td>AFFRI</td>
<td>7/74</td>
<td>AL</td>
<td>14</td>
<td>38.71</td>
</tr>
<tr>
<td>2486</td>
<td>378,546.142</td>
<td>1,961.829</td>
<td>AFFRI</td>
<td>7/74</td>
<td>AL</td>
<td>14</td>
<td>40.07</td>
</tr>
<tr>
<td>3071</td>
<td>378,546.142</td>
<td>1,961.829</td>
<td>AFFRI</td>
<td>7/74</td>
<td>AL</td>
<td>14</td>
<td>38.04</td>
</tr>
<tr>
<td>2602</td>
<td>463,356.822</td>
<td>2,340.749</td>
<td>AFFRI</td>
<td>7/74</td>
<td>AL</td>
<td>14</td>
<td>38.77</td>
</tr>
<tr>
<td>2259</td>
<td>563,981.102</td>
<td>2,768.589</td>
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<td>7/74</td>
<td>AL</td>
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<td>AFFRI</td>
<td>7/74</td>
<td>AL</td>
<td>14</td>
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<td>NTS</td>
<td>2/3/76</td>
<td>AL</td>
<td>14</td>
<td>35.97</td>
</tr>
<tr>
<td>3067</td>
<td>563,981.102</td>
<td>2,768.589</td>
<td>NTS</td>
<td>7/74</td>
<td>AL</td>
<td>14</td>
<td>37.06</td>
</tr>
<tr>
<td>3069</td>
<td>563,981.102</td>
<td>2,768.589</td>
<td>NTS</td>
<td>8/74</td>
<td>AL</td>
<td>14</td>
<td>37.88</td>
</tr>
</tbody>
</table>

¹ Fuel-follower control rod.

AFFRI = Armed Forces Fuels Research Institute.
AL = aluminum.
DORF = Diamond Ordnance Reactor Facility.
GA = General Atomics Company, Incorporated.
NRF = Neutron Radiography Facility.
NTS = Nevada Test Site.
SS = stainless steel.
TRIGA = Training, Research and Isotope Production, General Atomics.
The low burn-up has produced only a small amount of fission products. The relatively
long period since shutdown (May 1989) has allowed all of the halogens and most of the noble
gas inventory to decay.

The two FFCRs are irradiated and also have less than one percent burn-up. They were
last irradiated in 1976 and most of the high activity, short-lived fission products have decayed.

The source term was developed using ORIGEN2 code (WHC-SD-TP-ANAL-001). The
hypothetical maximum irradiated single fuel element radiation source term is 9.1 Ci, assuming a
minimum of six years decay from last irradiation. The total radionuclide inventory of
99 irradiated fuel elements and the two FFCRs is estimated to be a maximum of 920 Ci of
radioactive material, including less than 7.5 g of plutonium based on the maximum element. The
maximum total decay heat rate of the 99 fuel elements and 2 FFCRs is 2.7 W. The thermal
source for a single TRIGA fuel element is $2.51 \times 10^2$ W, with a design capacity of 18 elements
per cask yielding a total of 0.45 W per cask (WHC-SD-TP-ANAL-001, Part B, Chapter 8.0).
There was no previous operating history provided with the used fuel elements from the Armed
Forces Fuel Research Institute, Diamond Ordnance Reactor Facility, or the Nevada Test Site.
Operating history has been estimated and correlated to actual direct dose rate readings. Based on
this characterization effort, estimated exposures were normalized and ORIGEN2 runs generated
for all fuels (WHC-SD-TP-ANAL-001).

The TRIGA pool water quality was maintained with a purification system. Impurities
and minerals were removed to inhibit corrosion or filming. The purification system was
carefully monitored and recorded in a weekly log book. The pool water was sampled monthly
and tested by analytical chemistry to determine conductivity and pH balance and to verify that no
fission products were present in the water, demonstrating fuel cladding integrity.

**D2.5.1.3 Commercial Light Water Reactor Fuel.** Seven commercial LWR SNF assemblies
were delivered to the Pacific Northwest National Laboratory (PNNL) MCC at Hanford in 1985
in support of DOE spent fuel repository studies. Five were pressurized water reactor (PWR)
assemblies and two were boiling water reactor (BWR) assemblies. As part of the DOE spent fuel
repository studies, PNNL removed several fuel rods from the assemblies and designated them
Approved Testing Materials (ATMs) to support laboratory investigations of nuclear waste
disposal forms. As a result, the fuel from these assemblies has been well characterized.

Two of the five PWR assemblies supplied to PNNL contained three designations of spent
fuel from the Calvert Cliffs Unit No. 1 PWR operated by Baltimore Gas and Electric, located
near Lusby, Maryland. The first designation of fuel consisted of one full assembly containing
176 rods of moderate burn-up fuel and was identified as assembly D101. The second designation
consisted of a partial assembly containing 135 rods of high burn-up fuel and was identified as
assembly D047. Assembly D047 also contained twenty rods from a third designation of fuel that
was taken from assembly BT03 and inserted into assembly D047 to facilitate fuel shipment to the
PNNL facility.
For the PNNL MCC examinations, fuel rods taken from assemblies D101, D047, and BT03 were designated as ATM-103, ATM-104, and ATM-106, respectively. Assembly D101 had eight rods removed, leaving 168 rods remaining in the assembly. Six of the eight removed rods remain as intact loose rods for consolidation and shipping to storage. Assembly D047 had a total of sixteen rods removed, nine from assembly D047 fuel, and seven of the BT03 rods—leaving a total of 139 rods remaining in the assembly. Seven of the D047 rods and four of the BT03 rods remain intact for consolidation with the other loose rods for shipping and storage.

Fuel from the Point Beach assemblies was never used as sample material, consequently ATM designations were not assigned to this fuel and the assemblies have remained intact with none of the fuel rods removed. The three remaining PWR assemblies were supplied to PNNL from the Point Beach Unit No. 1 PWR operated by Wisconsin Electric Power Company in Wisconsin. These assemblies were received as three complete fuel assemblies, each containing 179 rods, with assembly identification numbers H-07, H-12, and H-25.

The segmented rod program (SRP) segmented rods from GE Vallecitos, which are actually quarter-length BWR fuel rods designed to screw together forming rod assemblies for irradiation in rod positions in standard BWR fuel assemblies, were irradiated in Quad Cities 1 (1-rod assembly) and Monticello (2-rod assemblies). Following destructive examination of six of these rods, one rod from Quad Cities (SRP-1) remains intact and five rods from Monticello (SRP-2) remain intact. Characterization data for the segmented rods includes axial core positions (i.e., top-center, bottom-center, or bottom) during irradiation. Table D2-10 provides a summary of the dimensions of the various fuel assemblies and fuel rods.

The BWR fuel rods are from two assemblies from Cooper Nuclear Power Plant, in Brownsville, Nebraska. These assemblies, CZ346 and CZ348, were designated as ATM-105, except for 10 rods (five from each assembly) that contain Gadolinia, which were designated as ATM-108. Twelve rods were removed from CZ346, of which three were destructively examined (including one ATM-108 rod) and nine loose rods remain intact. The remaining 37 rods from CZ346 and the 49 rods from CZ348 will be disassembled from the fuel bundle hardware for consolidation with the loose PWR and SRP rods in a container, which allows for handling and packaging comparable to a fuel assembly.
### Table D2-10. Commercial Light Water Reactor Fuel Physical Parameters. (2 sheets)

<table>
<thead>
<tr>
<th>Assembly No.</th>
<th>ATM No.</th>
<th>Manuf.</th>
<th>Bundle Length</th>
<th>Bundle Width</th>
<th>Rod Length</th>
<th>Pin Diameter OD, ID</th>
<th>Active Fuel Length</th>
<th>Pellet Length</th>
<th>Pellet Diameter</th>
</tr>
</thead>
<tbody>
<tr>
<td>D101 (Calvert Cliffs)</td>
<td>103</td>
<td>Combustion Engineering 14x14</td>
<td>157.25 in</td>
<td>8.125 in.</td>
<td>147 in.</td>
<td>0.388 in. ID 0.440 in. OD</td>
<td>136.7 in.</td>
<td>0.45 in.</td>
<td>0.3765 in.</td>
</tr>
<tr>
<td>D047 (Calvert Cliffs)</td>
<td>104</td>
<td>Combustion Engineering 14x14</td>
<td>157.25 in</td>
<td>8.125 in.</td>
<td>147 in.</td>
<td>0.388 in. ID 0.440 in. OD</td>
<td>136.7 in.</td>
<td>0.45 in.</td>
<td>0.3765 in.</td>
</tr>
<tr>
<td>BT03 (Calvert Cliffs)</td>
<td>106</td>
<td>Combustion Engineering 14x14</td>
<td>N/A</td>
<td>N/A</td>
<td>147 in.</td>
<td>0.388 in. ID 0.440 in. OD</td>
<td>136.7 in.</td>
<td>0.650 in.</td>
<td>0.3795 in.</td>
</tr>
<tr>
<td>H-07 (Point Beach)</td>
<td>N/A</td>
<td>Westinghouse 14x14</td>
<td>159.8 in.</td>
<td>7.76 in.</td>
<td>151.85 in.</td>
<td>0.422 in. OD</td>
<td>144 in.</td>
<td>0.600 in.</td>
<td>0.3659 in.</td>
</tr>
<tr>
<td>H-12 (Point Beach)</td>
<td>N/A</td>
<td>Westinghouse 14x14</td>
<td>159.8 in.</td>
<td>7.76 in.</td>
<td>151.85 in.</td>
<td>0.422 in. OD</td>
<td>144 in.</td>
<td>0.600 in.</td>
<td>0.3659 in.</td>
</tr>
<tr>
<td>H-25 (Point Beach)</td>
<td>N/A</td>
<td>Westinghouse 14x14</td>
<td>159.8 in.</td>
<td>7.76 in.</td>
<td>151.85 in.</td>
<td>0.422 in. OD</td>
<td>144 in.</td>
<td>0.600 in.</td>
<td>0.3659 in.</td>
</tr>
<tr>
<td>CZ 346 (Cooper Station)</td>
<td>105/108</td>
<td>General Electric 7x7</td>
<td>N/A</td>
<td>N/A</td>
<td>163.8 in.</td>
<td>0.563 in. OD 0.526 in. ID</td>
<td>146 in.</td>
<td>0.5 in.</td>
<td>0.477 in.</td>
</tr>
<tr>
<td>CZ 348 (Cooper Station)</td>
<td>108</td>
<td>General Electric 7x7</td>
<td>N/A</td>
<td>N/A</td>
<td>163.8 in.</td>
<td>0.563 in. OD 0.526 in. ID</td>
<td>146 in.</td>
<td>0.5 in.</td>
<td>0.477 in.</td>
</tr>
</tbody>
</table>

January 2000
<table>
<thead>
<tr>
<th>Assembly No.</th>
<th>Manuf.</th>
<th>Pin Diameter OD, ID</th>
<th>Rod Length</th>
<th>Bundle Length</th>
<th>Bundle Width</th>
<th>Active Fuel Length</th>
<th>Pellet Length</th>
<th>Pellet Diameter</th>
</tr>
</thead>
<tbody>
<tr>
<td>SRP-1 (Quad Cities)</td>
<td>N/A</td>
<td>N/A</td>
<td>N/A</td>
<td>N/A</td>
<td>N/A</td>
<td>N/A</td>
<td>N/A</td>
<td>N/A</td>
</tr>
<tr>
<td>Rod No 1803-4</td>
<td>General Electric</td>
<td>0.489 in.</td>
<td>38.5 in.</td>
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<td>N/A</td>
<td>N/A</td>
<td>0.42 in.</td>
<td>0.42 in.</td>
</tr>
<tr>
<td>SRP-2 (Montecello)</td>
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<td>38.41 in.</td>
<td>N/A</td>
<td>N/A</td>
<td>N/A</td>
<td>0.425 in.</td>
<td>0.425 in.</td>
</tr>
<tr>
<td>Rod No OA03-2</td>
<td>General Electric</td>
<td>0.463 in.</td>
<td>0.493 in.</td>
<td>38.41 in.</td>
<td>N/A</td>
<td>N/A</td>
<td>0.42 in.</td>
<td>0.42 in.</td>
</tr>
<tr>
<td>Rod No OA08-1</td>
<td>N/A</td>
<td>0.425 in.</td>
<td>0.493 in.</td>
<td>38.41 in.</td>
<td>N/A</td>
<td>N/A</td>
<td>0.42 in.</td>
<td>0.42 in.</td>
</tr>
<tr>
<td>Rod No SD17-3</td>
<td>N/A</td>
<td>0.425 in.</td>
<td>0.493 in.</td>
<td>38.41 in.</td>
<td>N/A</td>
<td>N/A</td>
<td>0.42 in.</td>
<td>0.42 in.</td>
</tr>
<tr>
<td>Rod No SD18-2</td>
<td>N/A</td>
<td>0.425 in.</td>
<td>0.493 in.</td>
<td>38.41 in.</td>
<td>N/A</td>
<td>N/A</td>
<td>0.42 in.</td>
<td>0.42 in.</td>
</tr>
<tr>
<td>Rod No 8D14-1</td>
<td>N/A</td>
<td>0.425 in.</td>
<td>0.493 in.</td>
<td>38.41 in.</td>
<td>N/A</td>
<td>N/A</td>
<td>0.42 in.</td>
<td>0.42 in.</td>
</tr>
</tbody>
</table>

ID = inner dimension, OD = outer dimension.
The five PWR fuel assemblies and two BWR fuel assemblies were delivered to Hanford and received by PNNL from October through December 1985. Subsequently, the assemblies, except for the rods removed for the ATM, have been stored in thimbles in air in a 324 Building hot cell. The removed rods were stored in a different hot cell during the time when the outer surface of the fuel assemblies became contaminated (primarily with cesium) from other hot cell activities. The contaminated assemblies will be spray-decontaminated with air and water prior to removal from the hot cell and subsequent cask loading for transfer to the 200 Area ISA.

The principal design parameters of the five PWR fuel assemblies, the two BWR assemblies from which the BWR rods are taken, and the SPR rod segments are described in this section. Table D2-11 also summarizes the assemblies and fuel rods, their fuel data, and cask designations.

Six NAC-1 spent fuel transport casks will provide on-site transport and interim storage of the spent fuel. Each of the five PWR fuel assemblies and a rod container consolidating the loose rods will be loaded into a seal-welded canister that will be placed within each NAC-1 cask. The following sections provide the principle design parameters of each fuel assembly by cask designation. The casks have been redesignated for the 200 Area ISA storage as follows:

- NAC-1A to NAC-1/ISA-A D101
- NSF-4B to NSF-4/ISA-B D047
- NAC-1C to NAC-1/ISA-C H-07
- NAC-1D to NAC-1/ISA-D H-12
- NAC-1E to NAC-1/ISA-E H-25
- NSF-4A to NSF-4/ISA-F Consolidated rods

PNL-11273, UC-812, Inventory of LWR Spent Nuclear Fuel in 324 Building, provides specific details concerning the fuel composition.

Cask NAC-1/ISA-A (Calvert Cliffs Fuel)

Cask NAC-1/ISA-A will contain fuel assembly D101. This assembly contains 176 fuel rods of moderate burnup fuel. Assembly D101 is a standard Combustion Engineering (C-E) 14x14 PWR fuel assembly. The assembly length is 157.25 in. The width of the fuel bundle is 8.125 in., and the fuel rods are 147 in. in length. All of the fuel rods are clad with Zircaloy-4 tubing. Additional fuel dimensional data are provided in Table D2-11.

The fuel rods from assembly D101 were irradiated during cycles 2, 3, and 4 of the Calvert Cliffs reactor operation between March 1977 and October 1980. The fuel assembly was discharged from the reactor October 18, 1980, and stored wet in the reactor fuel storage basin until September 1985, when it was shipped dry to PNNL. The fuel contains no burnable poisons and had an initial enrichment of 2.72 wt%. The initial total mass of uranium contained in assembly D101 with the actual remaining rods is 372.1 kg U. The burn-up of the assembly is 30,700 MWd/MTHM, and the calculated decay heat is 302.5 W.
Table D2-11. Commercial Light Water Reactor Fuel Rods.

<table>
<thead>
<tr>
<th>Cask no.</th>
<th>Fuel type</th>
<th>ID</th>
<th>No. rods</th>
<th>ATM no.</th>
<th>Burnup MWd/MTHM</th>
<th>Enrich wt. % (1)</th>
<th>Poison</th>
<th>Mass kg U</th>
<th>Manuf/ type</th>
<th>Clad</th>
<th>Disch date</th>
<th>Decay heat (watts)</th>
</tr>
</thead>
<tbody>
<tr>
<td>NAC-1/</td>
<td>PWR Assembly Calvert</td>
<td>D101</td>
<td>168</td>
<td>103</td>
<td>30,700</td>
<td>2.72</td>
<td></td>
<td>372.1</td>
<td>CE 14x14</td>
<td>Zircaloy-4</td>
<td>10/18/80</td>
<td>302.5</td>
</tr>
<tr>
<td>ISA-A</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>NSF-4/</td>
<td>PWR Assembly Calvert</td>
<td>D047</td>
<td>126</td>
<td>104</td>
<td>41,800</td>
<td>3.04</td>
<td></td>
<td>279.1</td>
<td>CE 14x14</td>
<td>Zircaloy-4</td>
<td>10/18/80</td>
<td>371.5</td>
</tr>
<tr>
<td>ISA-B</td>
<td></td>
<td>BT03</td>
<td>13</td>
<td>106</td>
<td>42,700</td>
<td>2.45</td>
<td></td>
<td>29.1</td>
<td>CE 14x14</td>
<td>Zircaloy-4</td>
<td>10/18/80</td>
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<td>NAC-1/</td>
<td>PWR Assembly Point</td>
<td>H-07</td>
<td>176</td>
<td></td>
<td>32,300</td>
<td>3.2</td>
<td></td>
<td>401.1</td>
<td>WE14x14</td>
<td>Zircaloy-4</td>
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<tr>
<td>ISA-C</td>
<td>Beach</td>
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<td></td>
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<td></td>
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<td>NAC-1/</td>
<td>PWR Assembly Point</td>
<td>H-12</td>
<td>176</td>
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<td>32,300</td>
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<td></td>
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<td>Zircaloy-4</td>
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<td>ISA-D</td>
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<tr>
<td>NAC-1/</td>
<td>PWR Assembly Point</td>
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<td>176</td>
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<td>32,300</td>
<td>3.2</td>
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<td>401.1</td>
<td>WE14x14</td>
<td>Zircaloy-4</td>
<td>10/19/81</td>
<td>356.4</td>
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<tr>
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<tr>
<td>NSF-4/</td>
<td>Consolidated Fuel Rods</td>
<td>D101</td>
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<td>13.29</td>
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<tr>
<td>ISA-F</td>
<td>PWR Rods Calvert Cliffs</td>
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<td>104</td>
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<td>15.50</td>
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<td></td>
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<td>BT03</td>
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<td>106</td>
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<td>2.45</td>
<td></td>
<td>8.96</td>
<td>CE 14x14</td>
<td>Zircaloy-4</td>
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<td>BWR Rods</td>
<td>CZ346</td>
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<td>143.9</td>
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<td>98</td>
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<td>2.5</td>
<td>Gd</td>
<td>35</td>
<td>GE 7x7</td>
<td>Zircaloy-2</td>
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<td>108</td>
<td>27,500</td>
<td>2.5</td>
<td></td>
<td>190.6</td>
<td>GE 7x7</td>
<td>Zircaloy-2</td>
<td>5/21/82</td>
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<td>105/1</td>
<td>22,600</td>
<td>2.56</td>
<td></td>
<td>0.63</td>
<td>GE Segm</td>
<td>Zircaloy-2</td>
<td>8/30/80</td>
<td>0.7 (3)</td>
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<td></td>
<td>Quad Cities 1</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
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<td></td>
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<td></td>
</tr>
<tr>
<td></td>
<td>BWR Seg Rods</td>
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<td>98</td>
<td>23,700†</td>
<td>2.87</td>
<td></td>
<td>3.17</td>
<td>GE Segm</td>
<td>Zircaloy-2</td>
<td>9/1/82</td>
<td>0.7 (3)</td>
</tr>
<tr>
<td></td>
<td>Monticello</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>316.4</td>
</tr>
</tbody>
</table>

(1) Average
(2) Initial enrichment
(3) Estimated from decay heat per BWR rod.

Information derived from:
WHC-SD-SNF-TA-010, 1996, Appendix A, Other Spent Nuclear Fuel Inventory Descriptions.
Cask NSF-4/ISA-B (Calvert Cliffs Fuel)

Cask NSF-4/ISA-B will contain fuel assembly D047. This assembly contains 126 original fuel rods and thirteen fuel rods from another assembly (BT03), for a total of 139 rods. Assembly D047 is a standard C-E 14x14 PWR fuel assembly. The assembly length is 157.25 in. The width of the fuel bundle is 8.125 in., and the fuel rods are 147 in. in length. All of the fuel rods are clad with Zircaloy-4 tubing. Additional fuel dimensional data are provided in Table D.2-11.

The fuel rods from assembly D047 were irradiated during cycles 2, 3, 4, and 5 of operation of the Calvert Cliffs reactor between March 1977 and April 1982. The fuel assembly was discharged from the reactor April 17, 1982, and stored wet in the reactor fuel storage basin until September 1985, when it was shipped dry to PNNL. This fuel contains no burnable poisons and had an initial enrichment of 3.04 wt%. The initial total mass of uranium contained in the remaining D047 fuel rods (excluding the BT03 rods) is 279.1 kg U. The burn-up of the assembly is 41,800 MWd/MTHM, and the calculated decay heat is 336.7 W.

The fuel rods from assembly BT03 were irradiated during cycles 1, 2, 3, and 4 between October 1974 and October 1980. The fuel assembly was discharged from the reactor October 18, 1980. Thirteen fuel rods from this assembly are contained within assembly D047 and will be stored within cask NAC-1-LWR2-D047. This fuel contains no burnable poisons and had an initial enrichment 2.45 wt%. The initial total mass of uranium contained in the thirteen BT03 fuel rods is 29.2 kg U. The burn-up of the fuel is 42,700 MWd/MTHM, and the calculated decay heat for the 13 remaining fuel rods is 34.7 W.

The D047 fuel rods combined with the BT03 fuel rods in the D047 assembly result in 308.2 kg U for the total mass of uranium contained within the assembly. The combined heat load for the assembly results in 371.5 W. This value is the bounding heat load expected from any of the PWR assemblies.

Cask NAC-1/ISA-C (Point Beach Fuel)

Cask NAC-1/ISA-C will contain fuel assembly H-07. This assembly contains 176 fuel rods. Assembly H-07 is a standard Westinghouse 14x14 design. The assembly length is 159.8 in. The width of the fuel bundle is 7.76 in., and the fuel rods are 151.85 in. in length. All of the fuel rods are clad with Zircaloy-4 tubing. Additional fuel dimensional data are provided in Table D2-11.

The fuel rods from assembly H-07 were irradiated during cycles 5, 6, 7, 8, and 9 between November 1976 and October 1981. The assembly was discharged from the reactor October 9, 1981. This fuel contains no burnable poisons and has an initial enrichment of 3.19 wt%. The
initial total mass of uranium contained in the assembly is 401.1 Kg U. The burn-up for assembly H-07 is 32,300 MWD/MTHM, and the calculated decay heat is 356.4 W.

Cask NAC-1/ISA-D (Point Beach Fuel)

Cask NAC-1/ISA-D will contain fuel assembly H-12. This assembly is identical in fuel description and irradiation history to assembly H-07. This fuel has an initial enrichment of 3.19 wt%. The initial total mass of uranium contained in the assembly is 401.1 kg U. The burn-up of the fuel is 32,300 MWD/MTHM, and the calculated decay heat is 356.4 W.

Cask NAC-1/ISA-E (Point Beach Fuel)

Cask NAC-1/ISA-E will contain fuel assembly H-25. This assembly is identical in fuel description and irradiation history to assemblies H-07 and H-12. This fuel has an initial enrichment of 3.19 wt%. The initial total mass of uranium contained in the assembly is 401.1 kg U. The burn-up of the fuel is 32,300 MWD/MTHM, and the calculated decay heat is 356.4 W.

The BT03 fuel contained within the D047 assembly is the most recent fuel of the five PWR assemblies to have been irradiated in a reactor. The BT03 rods were last irradiated in 1982. The remaining PWR fuel was discharged from the reactor before this date. Therefore, a minimum acceptable cooling time after removal from the reactor, before storage of the PWR assemblies at the ISA, is established at 14 years. During this storage period, unburned uranium and fission by-products from reactor irradiation of the fuel have continued to radioactively decay.

Cask NSF-4/ISA-F (Cooper, Calvert Cliffs, Quad Cities I, and Monticello Fuel)

Cask NSF-4/ISA-F will contain 95 163.8-in. rods from two standard, GE 7 x 7 BWR assemblies, CZ346 and CZ348; six 147-in. rods from D101, seven 147-in. rods from D047; four 147-in. rods from BT03; one 38-in. GE BWR rod from SRP-1; and five 38-in. GE BWR rods from SRP-2. The D101, D047, and BT03 fuel and irradiation history are described with the contents of Cask NAC-1/ISA-A and Cask NSF-4/ISA-B. Additional fuel dimensional data are provided in Table D2-11.

Fuel assembly CZ346 and CZ348 were irradiated in cycles 1, 2, 3, 6, and 7 of the operation of the Copper Nuclear Power Plant between July 1974 and May 1982. The two assemblies were shipped from the Cooper storage pool to the pool at the GE Morris Facility, used in a dry storage cask test, and shipped to Pacific Northwest Laboratory in January 1986. No adverse effects on fuel rod integrity were found after testing. Nine of the rods contain Gadolinia. The initial enrichment of the fuel was 2.5 wt%. The initial total mass of uranium in the remaining rods is 370 kg U. The burn-up of the fuel is 28,000 MWD/MTHM, and the calculated decay heat is 272 W.
The irradiation history and shipments to PNNL for the PWR rods is included in the descriptions in the above paragraphs. The SRP-1 rods from Quad Cities I reactor were irradiated during cycles 2 through 5, between July 1974 and August 1980, for a burnup of 22,600 MWd/MTHM. SRP-2 rods from Monticello reactor were irradiated during cycles 3 through 8 (three rods) and 3 through 9 (two rods), between May 1974 and April 1981 and between May 1974 and September 1982, respectively. The average burnup is 23,700 MWd/MTHM. The initial total mass of uranium in the SRP-1 is 0.81 kg U. and 3.17 kg U in the five SRP-2 rods. The decay heat of the SRP rods is estimated to be 4.3 W. The combined heat load for this cask is 316.4 W.

D2.6 CONFINEMENT SYSTEMS

10 CFR 72.122(h)(1) requires that (1) confinement barriers and systems be provided such that the spent fuel cladding is protected during storage against degradation that leads to gross ruptures of the fuel, or (2) the fuel must be otherwise confined such that degradation of the fuel during storage does not pose operational safety problems with respect to its removal from storage. This section identifies and describes the functional, physical, and operational features of the confinement systems that provide protection against release of radioactive or hazardous material. Detailed system descriptions are provided in Chapter D4.0.

D2.6.1 Confinement Approach and Configuration

The primary confinement feature in the three different types of dry cask storage systems is the intact fuel cladding. The inner or outer storage container provides a second confinement boundary. Specific details for each storage system are provided in the subsections that follow.

D2.6.2 Confinement System Descriptions

D2.6.2.1 Fast Flux Test Facility Fuel. The primary confinement feature is the intact fuel cladding, which serves to ensure cladding equivalency is maintained during the spent fuel dry storage design lifetime. The secondary confinement feature consists of the ISC, which serves as the main pressure boundary and leakage barrier to the environment to ensure that environmental, worker, and public safety is maintained. The ISC design is consistent with commercial spent fuel dry storage practice. This design restricts oxygen and moisture in-leakage to minute levels such that degradation of the spent fuel is expected to be effectively controlled to an inconsequential rate over the 50-year storage lifetime. The CCC provides fuel retrieval capability, if degradation does occur.
Interim Storage Cask Confinement Boundary

The ISC confinement structure consists of the 1.5-in. thick annular wall, an 8-in. thick bottom head, an 8-in. thick top closure plate, a 4-in. thick top flange, two test-port welded covers, and double metal seals between the top closure and the flange. The top closure of the ISC is held in place by sixteen 1.5-in. diameter bolts. The test port through the closure is provided for evacuation and helium backfill of the cask, once the CCC is installed. After the cask is inerted with helium and the main closure seals certified to a leak rate of \( <1 \times 10^{-7} \text{scc/sec} \), the two 0.5-in. thick test port covers are sequentially seal welded in place and dye penetrant tested. The test port also allows sampling of the cask atmosphere, but no such sampling is currently required or anticipated during normal storage operations. The confinement barrier design conforms to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME 1995, Section III, Subsection NC, for safety-significant components. Guidance for application of the ASME Code was taken from NRC Regulatory Guide 7.6, Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels.

Core Component Container Boundary

The CCC is constructed to "can" the spent fuel and provide a barrier, in addition to the intact FFTF spent fuel cladding. This barrier is provided to ensure retrievability of the spent fuel in the unlikely event of significant cladding degradation. The CCC confinement boundary consists of the seven closed-bottom storage tubes welded to a single upper support plate, which provides the sealing surface, and the bolted cover that has a metallic seal (see Figure D2-9). The metallic seal between the cover and the container is positioned to pass radially inward of the bolt holes and outward of the upper rim of the storage tubes, thus excluding the closure bolts from this boundary. Additional elastomer seals are located at the outer circumference of the outer plate and at each bolt. A 0.375-in. diameter test port is provided to access the space between the metal and elastomer seals for leak testing. Before acceptance from the manufacturer, each CCC was certified to a leak rate of \( <1 \times 10^{-3} \text{scc/sec} \). After loading the CCC with spent fuel in the interim examination and maintenance (IEM) cell, the seal leak rate was verified at \( <1 \times 10^{-1} \text{scc/sec} \). The IEM cell atmosphere was the inerting atmosphere for long-term storage of the spent fuel because the CCC was closed prior to transfer out of the IEM cell. This atmosphere is argon with minor (<200 ppm) levels of oxygen and water impurities. The potential levels of oxygen and water within the IEM cell were evaluated in WHC-SD-FF-TA-039, Radiological Consequences of a Hypothetical Disruption of a Maximally Loaded FFTF Fuel Cask, and determined to have no significant effect on the spent fuel storage. Maintaining an argon atmosphere is not a safety function.

The CCC (Figure D2-9) is an unshielded, sealed fuel storage container with seven fuel storage positions. The CCC provides canning for the fuel to ensure fuel retrievability is maintained during the spent fuel dry storage design lifetime. The CCC also provides the geometry to ensure criticality control during handling and storage of the fuel. The CCC is
designed such that it is fully retrievable from the storage configuration, although the capability to remove individual fuel components from a CCC is not required or guaranteed. The center storage location can accept a fuel assembly that has had the bottom 15.5 in. removed. However, due to the indented CCC handling socket, an Ident-69 pin container cannot be stored in the center location. Therefore, the maximum loading of a CCC will be either six Ident-69 pin containers or seven FFTF fuel assemblies. Pin containers and fuel assemblies can be mixed in a CCC with up to five Ident-69 containers and two DFAs.

The CCC is fabricated from stainless steel and nickel alloy material to provide corrosion-resistant fuel storage. The CCC's overall dimensions are 20.0 in. in diameter by 146 in. in height. The weight of the empty CCC is 1,100 lbs. The maximum gross weight of a loaded CCC occurs with seven DFAs. The weight of the seven DFAs is 3,900 lbs, giving the CCC a maximum gross weight of 5,000 lbs.

The upper portion of the outer tubes has a 6.69-in. outside diameter. There is a saddle section 14 in. above the bottom of the CCC, where each outer storage tube transitions to a smaller section measuring 4.00 in. in outside diameter. The bottom 12.4 in. of the lower section is fabricated from nickel alloy material for enhanced corrosion resistance. The center tube has a 6.54-in. outer diameter. The bottom 10.0 in. of the center tube is also fabricated from nickel alloy material. There is no size reduction at the lower end of the center tube.

The outer storage tubes are suspended from the upper support plate. The center storage tube connects the upper support plate with the lower support plate. The outer tubes are fixed only at their upper ends so they are free to accommodate thermal expansion. This design also permits the outer tubes to stretch slightly to absorb energy during a CCC drop accident onto the ISC internal impact limiter. Drop energy is absorbed by the CCC tubes until the gap between the tube stop and the lower support plate is taken up. The CCC then acts as a rigid body for final interface with the ISC impact limiter during an accident event.

The lower support plate has an 18.0-in. diameter and is 1.5 in. thick. This support plate limits downward travel of the outer storage tubes, and also provides radial positioning guidance for inserting the CCC into the ISC. The upper support plate has a 20.0-in. diameter, with an overall thickness of 3.6 in. It provides support for the outer storage tubes and the seating surface for the cover seal. There are twelve drilled holes in the upper support plate that accommodate the cover bolts.

The cover has a 20.0-in. diameter and an overall thickness of 1.63 in. The lower surface of the handling socket extends 8.35 in. below the bottom of the cover. Twelve holes are drilled in the cover and are sized to accommodate the closure bolts. The CCC atmosphere is free to pass between the storage tubes, but a metal Helicoflex\(^1\) seal is provided between the container and the

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\(^1\) Helicoflex is a trademark of Cefilac, Societe Anonyme.
cover to establish container confinement. An impact limiter is located on the bottom of the handling socket closure cup. The impact limiter function is to reduce the loading to the cover that could occur during the CCC drop accident into the ISC due to the fuel assembly in the center storage location rebounding and hitting the cover. The top surface of the cover is provided with rigging attachment points for empty container handling, a test port for seal leak rate verification, and a sample port for container atmosphere sampling.

The CCC cover design provides a closure with a cover that mates to the CCC body, crushing a metallic O-ring between them, which provides confinement for the spent fuel. Leak testing requirements assure that the CCC will function as an effective "canning" barrier. After the CCC is loaded with spent fuel in the IEM cell, a pressure decay test to \( \leq 1 \times 10^{-1} \text{ scc/sec} \) is performed on the seal to verify correct installation of the metal Helicoflex seal. Additionally, each fuel storage tube is closed at the bottom with a nickel alloy cup to prevent potential caustic fission product solutions from degrading the ISC confinement liner in the unlikely event of a leak out of a fuel pin. Additional features of the CCC that provide "canning" assurance are: (1) all pressure boundary welds are required to be full penetration and dye penetrant inspected during the fabrication process, and (2) each CCC is hydrostatically tested to 105 lbs/in\(^2\) gauge for the design pressure of 70 lbs/in\(^2\) gauge, with a pressure of 62 lbs/in\(^2\) gauge resulting from 100% fission gas release of seven fuel assemblies.

**D2.6.2.1.1 Primary Confinement.** The FFTF fuel cladding is intact and credited as the primary confinement boundary, thus meeting the intact fuel criteria when the fuel is placed into the CCC. Cladding integrity is verified when each spent fuel assembly is washed using the sodium removal process in the IEM cell. A quantitative assessment of cladding integrity is performed before transfer to dry storage; in addition, contamination of the sodium removal wash water provides an indication of a gross cladding failure during this process.

**D2.6.2.1.2 Secondary Confinement.**

**Interim Storage Cask**

The ISC is an aboveground concrete and steel shielded, top-loading spent fuel storage cask that will be used to provide safe interim dry storage of a CCC with FFTF spent fuel assemblies or pin containers for a period of up to 50 years. One CCC can be stored in the cavity of each ISC. The ISC design consists of an all-stainless steel internal confinement structure surrounded by steel and concrete shielding. The fully loaded cask weighs a maximum of 114,200 lbs, including a loaded CCC with a gross payload of 5,000 lbs, the closure hardware, and the weather cover. Outer cask dimensions are 85 in. in diameter and 181 in. tall. The internal cavity of the ISC is 21 in. in diameter and 147 in. tall, and will accept one CCC. This cavity, which is formed by a 1.5-in. thick stainless steel cylinder and 8-in. thick top and bottom closure plates, provides the confinement boundary.
The ISC has been designed and fabricated to meet the requirements of WHC-S-4100, in accordance with 10 CFR 72. The ISC is designed to provide confinement for the fuel, passive heat removal, and environmental protection for the CCC. It also provides radiological shielding protection for site personnel by limiting the dose rate at normally accessible surfaces to acceptable levels. A gasketed weather protection cover is installed on each ISC in the ISA. An additional cover plate may be seal welded over the bolted closure plug after receipt at the 200 Area ISA to enhance the long-term storage configuration.

The ISC confinement boundary design and analysis were performed by General Atomics (1995), which holds an ASME Certificate of Authorization N for design and overall fabrication responsibility for ASME Code, Section III, Division 1 and 2 components. The ISC design analysis and material properties were based on the requirements of the ASME Code, Section III, Subsection NC. Fabrication of the ISC is based on ASME Code, Section III, Article NC-4000. The ISC confinement boundary is constructed of ASTM materials. Additionally, all ISC design and fabrication activities were required to be performed using a 10 CFR 72 quality assurance program or equivalent.

The bottom of the ISC cavity is fitted with an aluminum crush pad to limit CCC impact loads in the unlikely event that it is dropped into the ISC during loading. The surfaces of the cavity are finished to remove burrs, sharp corners, and weld beads that potentially could interfere with cask loading operations. The confinement structure described above also is surrounded by annular steel shield plates that are surrounded by concrete reinforced with rebar. As discussed in the FFTF Spent Fuel Interim Storage Cask Design Analysis Report (General Atomics 1995), the concrete shielding structure is designed to meet ACI-349, Code Requirements for Nuclear Safety Related Concrete Structures. The concrete design mix was selected to ensure strength and long life over the range of temperature conditions expected during normal operations and the more extreme short-term temperatures that could occur during off-normal or accident conditions.

The ISC heat dissipation is totally passive. All required heat removal will occur by conduction, thermal radiation, and convective cooling of the outer surface. The ISC also is provided with a passive cooling system that removes heat by using an internal natural circulation airflow system. The airflow system is formed by two inlet ducts, an annular gap between the confinement boundary cylinder and the inner shield, and two outlet ducts. The inlet and outlet ducts are steel-lined penetrations through the concrete that take non-planar paths to minimize radiation streaming. These penetrations consist of two 4-in. outer diameter ducts that supply air to the bottom of a 0.75-in. wide annulus gap between the confinement boundary cylinder and the inner shielding cylinder. Natural convection circulates air up the annular space and out two similar ducts at the top of the annulus.

The ISC provides several layers of shielding to ensure worker ALARA (as low as reasonably achievable) radiation protection. This protection is based on the requirement that the 200 Area ISA is unmanned. The first layer of shielding consists of a 3.0-in. thick carbon steel
clamshell around the entire length of the cavity. Another partial length, 4.0-in. thick, carbon steel clamshell shield is provided for additional shielding at the cavity mid-plane where the fuel section is located. The clamshell shields are designed with studs that attach to the reinforced concrete cylinder shield. The concrete is a minimum of 21.25 in. thick for additional radial shielding. Supplemental axial shielding is provided by 4.0-in. thick plates below the bottom head and below the upper closure. A design dose rate of 2 mrem/hr at accessible surfaces for normal conditions and 200 mrem/hr at the (inaccessible) bottom head is defined in WHC-S-4100. Shielding acceptance criteria in the design specification allowed a localized dose rate of 5.0 mrem/hr to account for potential shielding imperfections and localized hot spots.

D2.6.2.2 NRF TRIGA Fuel. A Rad-Vault is used to store NRF TRIGA casks and DOT-6M containers, which hold TRIGA SNF from the NRF in the 300 Area. Each NRF TRIGA cask can hold up to 18 elements and is nominally 38 in. tall and 16 in. in diameter. The weight of the outer vessel is approximately 1,637 lbs. The entire NRF TRIGA cask weighs a total of 2,013 lbs when filled with 18 elements (WHC-SD-TP-SARP-008). Each DOT-6M container holding one FFCR in an inner 2R container is nominally 70 in. tall and 23 in. in diameter. The weight of the loaded DOT-6M container is approximately 640 lbs. The Rad-Vault is a concrete right circular cylinder with light steel reinforcement. The empty Rad-Vault weight is 63,400 lbs (43,400-lbs body and 20,000-lbs lid) and the maximum design weight with payload is 81,400 lbs; however, the maximum expected weight is approximately 77,000 lbs. With the lid installed, the Rad-Vault is 111 in. tall, with an outer diameter of 114 in.

The concrete Rad-Vault is a low-level radioactive storage unit. The Rad-Vault will be placed on compacted gravel at the ISA, and top loaded with the NRF TRIGA casks and DOT-6M containers. The Rad-Vault is equipped with a removable lid that fully exposes the available internal storage volume. The lids' mating surface to the main container is sloped to prevent rain intrusion. Opposing lifting lugs are interlaced into the steel rebar and welded wire fabric, and cast into the concrete structure. Lift capacity is sufficient to allow an empty Rad-Vault to be lifted and moved by crane to a transporter. The lid must be transported separately. The Rad-Vault is not designed to be lifted loaded or with the lid installed.

Sampling and/or drain capability is provided through a siphon-type arrangement that prevents leakage in and out of the Rad-Vault. Each Rad-Vault is also equipped with a pop-up vent. No gas is expected to be vented; however, the pop-up vent and siphon will provide a vented system. There are no safety requirements for the vent or the drain functions.

NRF TRIGA Cask/DOT-6M Container Confinement Boundary

The outer container of the NRF TRIGA cask is the qualified and tested confinement barrier for storage. This container has a Helicoflex metallic seal for long-term storage integrity, an elastomeric flange seal to facilitate leak testing, and an elastomeric bore seal. The quick-disconnects used for leak testing are also fitted with metallic O-ring seals at the thread interface.
The port lids covering the quick-disconnects used for helium leak testing are also provided with single metallic Helicoflex seals.

The 2R inner container of the DOT-6M system is the qualified and tested confinement barrier. This 5-in. diameter pipe vessel has a bolted flange assembly retaining an elastomeric seal and a Helicoflex metallic seal; helium leak test capability is included.

Helicoflex seals are fabricated using an inner spring covered by an inner lining and a soft outer lining. The Helicoflex seal for the NRF TRIGA and DOT-6M has a Nimonic 90 Spring\(^4\) covered by an inner lining of Inconel\(^5\) Alloy 600 and an outer jacket of aluminum. The materials used in the linings are application-dependent and are a function of temperature and pressure. The sealing principle of Helicoflex is based, in part, on plastic deformation of the seal’s outer lining. The inner helical spring provides an elastic core under bolt preload. Each coil of the spring behaves independently during radial compression. The all-metal design is the reason for its long life.

The Rad-Vault provides shielding, safeguards, and weather protection, but it is not a sealed system and is not designed to provide confinement.

The NRF TRIGA cask provides multiple barriers. The inner container has an elastomeric seal on the lid and a metallic seal on the clamping eye bolt. The outer container has redundant seals provided by both an elastomeric bore seal O-ring and a Helicoflex combination metallic/elastomeric O-ring seal. Both the bore seal and the Helicoflex seal are leak tested before transport and/or storage.

The closure devices on the NRF TRIGA cask include the closure lid and the three leak test port lids. The closure lid is attached to the cask body with twelve 0.50-in. diameter cap screws. All leak test port lids are attached to the cask lid with six 0.25-in. diameter cap screws.

The NRF TRIGA cask is provided with removable test port covers that permit access to the helium leak test components in the outer container. The container is leak tested upon initial loading according to the standards of American National Standards Institute (ANSI) N14.5, American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment, for a maximum leak rate of $1.0 \times 10^{-7}$ scc/s (air), with the exception that the test port covers are tested to $1 \times 10^{-3}$ scc/s (air). No additional testing is anticipated or required.

\(^4\) Nimonic 90 Spring is a trademark of Inco Alloys International, Inc.

\(^5\) Inconel is a trademark of Inco Alloys International, Inc.
The DOT-6M containers used for storage of the TRIGA FFCRs provide multiple confinement barriers. The intact cladding is a barrier. The 2R inner container provides a confinement barrier with the combination metallic and elastomeric O-ring seals.

The DOT-6M container’s outer stainless steel drum can be opened to permit access to the helium leak test port on top of the 2R inner container. The 2R inner container within the DOT-6M is secured by eight 0.75-in. diameter bolts. This vessel is leak tested in accordance with ANSI N14.5 standards, for a maximum leak rate of $1.0 \times 10^{-7}$ scc/s (air).

**D2.6.2.2.1 Primary Confinement.**

### NRF TRIGA Cask

For the NRF TRIGA cask, the intact cladding is the primary barrier. In order to demonstrate fuel cladding integrity, the TRIGA pool water quality was maintained with a purification system. Impurities and minerals were removed to inhibit corrosion or filming. The purification system was carefully monitored and recorded in a weekly log book. The pool water was sampled monthly and tested by analytical chemistry to determine conductivity and pH balance and to verify that no fissionable gases were present in the water, thereby demonstrating fuel cladding integrity.

### DOT-6M Container

For the DOT-6M system, the intact cladding is the primary barrier. As noted above, the TRIGA pool water quality was maintained with a purification system. Impurities and minerals were removed to inhibit corrosion or filming. The purification system was carefully monitored and recorded in a weekly log book. The pool water was sampled monthly and tested by analytical chemistry to determine conductivity and pH balance and to verify that no fissionable gases were present in the water, thereby demonstrating fuel cladding integrity.

**D2.6.2.2.2 Secondary Confinement.**

### NRF TRIGA Cask

The inner container has an elastomeric seal on the lid and a metallic seal on the clamping eye bolt. No credit is taken for confinement provided by this barrier.

The outer container of the TRIGA cask provides confinement and has redundant seals provided by both an elastomeric bore seal O-ring and a Helicoflex combination metallic-Viton O-ring seal. Both the bore seal and the Helicoflex seal are leak tested before transport and storage. Leak test port covers have a single Helicoflex metallic seal that is also leak tested.
DOT-6M Container

The 2R inner container of the DOT-6M is a qualified and tested confinement barrier. This 5-in. pipe vessel has a bolted flange assembly retaining an elastomeric seal and a Helicoflex metallic seal; helium leak test capability is included.

The stainless steel 6M drum provides a barrier for normal conditions only. This barrier contains pressure relief capability via the penetrations covered with pressure-sensitive adhesive filament (WHC-S-0393). This drum does not provide a safety-related confinement function, but acts as an impact absorber for the 2R container.

D2.6.2.3 Commercial Light Water Reactor Fuel. The commercial LWR fuel storage system consists of six storage units, each comprised of a canister, a NAC-1 cask (Figure D2-10 and D2-11), and an ISO shipping/storage container (Figure D2-12 and D2-13). Each canister will contain a single PWR fuel assembly, or in the case of the consolidated loose rods, a rod container (basket) designed to maintain the positioning upon which the criticality safety analysis is based. The canister provides the secondary containment, as required by 10 CFR 72. Analysis shows that damage to the canister is not credible. The canister will be placed within the NAC-1 cask, which will provide shielding and structural protection of the fuel during storage.

The NAC-1 casks, previously licensed by the NRC to transport LWR spent fuel and waste material, will be modified for use at Hanford for both transportation and storage purposes. The NRC license will not be retained. Modifications to the cask include the removal or plugging of several valves connected to the confinement cavity. The NAC-1 casks are mounted to supports within the ISO container for transportation and will remain in the containers in this configuration during storage at the 200 Area ISA.

The storage units will be placed by crane alongside each other (1 x 6 array) and evenly spaced 4 ft. apart. This spacing is not required for criticality or other safety analysis purposes, but rather for personnel access considerations and maintaining personnel radiation doses ALARA, since this is an unmanned facility.

Light Water Reactor Canister Description

The canisters are based on SNF Project acceptance criteria for the LWR fuel dry cask storage system at the 200 Area ISA (SNF-4894). These criteria include the specification that the design and fabrication meet the requirements of ASME Code, Section III, where necessary to maintain geometry control such that 0.95 k$_{\text{eff}}$ is not exceeded. The design life expectancy of the canisters is 40 years.

The canister provides the secondary confinement boundary for the fuel. The canister is fabricated from a 12-in. schedule 40 stainless steel pipe, with a welded base cap and top closure.
It has a maximum outside diameter of 13.0 in., including the end cap, and a total length of 165.25 in. The bottom cap is machined, so the pipe-to-cap weld is inspectable at the side of the canister. The top closure lid weld will be tested for leak tightness per the requirements of ANSI N14.5. The closure lid contains a penetration that allows the canister to be evacuated for vacuum drying of the fuel assemblies, filled with helium, leak tested, and welded closed. The canister end cap allows the installation of handling attachments for cask loading and subsequent transloading operations. Figure D2-14 provides an illustration of the canister.

The canisters are provided with safety class internal supports for the fuel assemblies or the loose rod container, which provide geometry control such that a loaded canister will have a $k_{\text{eff}} < 0.95$ when fully moderated and reflected under all credible accident conditions. Gamma shield plugs will be provided to be placed in the cask cavity under and on top of the canister.

The maximum loaded weight of the canister is not to exceed 3,300 lbs. The maximum internal design pressure of the canister is 50 lbs/in$^2$ gauge testable to a pressure of 85 lbs/in$^2$ gauge.

The canister for the consolidated loose rods requires the rods be provided with safety-class containment within the canister so that criticality limits are not exceeded. The $k_{\text{eff}}$ of the optimum pitch of the "bundle" of fuel rods in a canister with full water moderation and reflection was calculated to be $>0.95$ and $<0.97$. To limit $k_{\text{eff}}$ to $<0.95$, the diameter of the bundle of rods must be limited to 10.9 in. The design shall provide for this requirement. A bounding design solution for purposes of providing for this requirement in a manner conservative for calculations of pressure releases, a 10-in. safety-class pipe sleeve is installed into the canister to back up the rod container. The rod container is proposed to be an 8-in. schedule 80 diameter pipe that is split to accommodate rod loading, after which it is closed and fastened prior to loading into the canister. Figures D2-14 and D2-15 depict this approach. The rod container and the backup pipe sleeve provide for helium flow to be in contact with the fuel rods for vacuum drying and helium backfill.

**Cask Description**

The acceptance criteria require that the NAC-1 cask is designed and fabricated to the requirements of the ASME Code, Section III (1971). The external shape of the NAC-1 spent fuel shipping cask approximates a smooth-surface, right circular cylinder that is modified, in that impact limiters protrude radially at both ends. The internal cross-section of the cask cavity is circular. The overall dimensions of the cask include a length of 214 in. (including lid impact limiter) and a maximum cross-sectional envelope diameter of 50 in. The internal cavity of the cask is 178 in. long and 13.5 in. in diameter. The maximum loaded gross weight of the cask, including the maximum fuel and canister weight (3,300 lbs.), is approximately 49,000 lbs. The principle design features of the cask are the transportation confinement boundary, shielding and heat dissipation systems, and the lifting and tie-down systems.
The structures of the NAC-1 cask are constructed of stainless steel. The cask cavity is formed by the inner shell, which is a 14.125-in. outside diameter, 0.3125-in. thick stainless steel shell. The upper end of the shell is welded to the cask cavity flange; the bottom end of the shell is welded to the cask bottom casting. Surrounding the inner shell of the cask is a nominal 6.6875-in. thick annulus of chemical grade lead (gamma) shielding. The lead is shaped such that approximately 5 in. from the bottom and 30 in. from the top, the thickness is reduced to 5.4375 in. There is an annular void, 5 in. long by 1.25 in. thick, at the bottom end of the gamma shield to allow for any lead expansion during the fire accident. The upper axial shaping is accomplished by reducing the diameter of the outer shell 2.50 in. over a 30-in. length. The lead/steel interface of the inner and outer shell have axial copper fins that are imbedded in the lead and welded to the inner and outer shells to transfer heat across the interface with a minimum temperature gradient.

The confinement boundary of the NAC-1 cask is the inner shell, lower end casting, upper end casting, bolted closure lid with double neoprene O-ring seals and seal test port, a helium fill/vent valve, rupture disks, and lower casting drain valves. Because the welded canister provides the final confinement boundary for storage, it will not be required to inspect and replace the O-ring seals. The NAC-1 cask will be modified to accept the designed canister and further minimize worker exposure by eliminating unnecessary surveillance and maintenance activities during storage at the 200 Area ISA. The modifications, in accordance with SNF-4894, Spent Nuclear Fuel Project Acceptance Criteria for LWR Fuel Storage System, consist of removing the anti-rotational lugs in the cask cavity and replacing drain, relief, and fill valve penetrations and rupture disks with pipe plugs. The cover plates will be reinstalled over the ports, as designed, to serve as heat shields during the postulated design basis fire accident. Also, the neutron shield tank burst disk assembly port penetrations will have the existing assembly nut, rupture disk, and muffler removed. Solid plugs will be installed in place of the assembly nuts.

The outer shell is formed by a 30-in. diameter, 1.25-in. thick stainless steel cylinder reduced to a 27.50-in. diameter at one end. The cask bottom consists of a shaped stainless steel disc with a 30-in. outer dimension, and an 8-in. thickness that functions as a gamma shield for the bottom end of the cask. The cavity flange is a stainless steel ring with a 29.75-in. outer dimension, 17.50-in. inner dimension, and an 8.625-in. thickness. The bottom disc end and top flange are welded to the inner and outer shells to form the enclosure for the lead gamma shield.

Neutron shielding tanks are provided in the 4.50-in. thick annular space formed between the outer shell of the lead gamma shield and a thin stainless steel shell that constitutes the outer cask surface. The neutron shield tanks will not be used for the storage configuration at the 200 Area ISA and will contain air.

Upper end shielding is provided by the 7.50-in. thick stainless steel cask lid. The cask lid is a stainless steel casting that also serves as a gamma shield. The lid is a flanged frustum of a cone, 7.50 in. thick with a maximum diameter of 25.50 in. The conical portion of the stainless
The flanged portion of the lid is a 2-in. thick, 25.50-in. diameter disc. There are six counter-bored clearance holes for the lid bolts and four 1-in. diameter blind-threaded holes for attaching the lid impact limiter. The cask lid is bolted to the cavity flange by six 1.25-in. diameter hex-head bolts. Bolt heads bear on the cask lid; the shanks penetrate through the lid flange and thread into the cavity flange. Six 1.25-in. diameter holes with HeliCoil\textsuperscript{\textregistered} thread inserts are provided in the cavity flange for bolting the cask lid to the flange. The bolt heads are drilled for a wire security seal.

The lower end impact limiter structure is a ring that surrounds the cask lower casting, formed from a stainless steel sheet and/or plate welded to the cask outer shell and flange areas. The impact limiter was designed to absorb the energy of the design basis 30-ft. free drop. It contains a balsa wood disc placed adjacent to the cask bottom. A 0.125-in. thick sheet of asbestos is placed between the balsa and the cask bottom, and is contained internally. Within the impact limiter, extending radially from the center of the cask, are eight 0.375-in. thick stainless steel gussets. The bottom section of the impact limiter also functions as a pedestal for supporting the cask in the vertical position.

An impact limiter is located at the upper end of the cask body, designed to absorb the energy of the design basis side drop accident. The upper impact limiter is a stainless steel-sheathed, balsa-filled ring that surrounds the cask cavity flange. A 0.125-in. sheet of asbestos is positioned between the balsa and cask outer shell, and is contained internally. Within the impact limiter, extending radially from the center of the cask, are eight 0.375-in. thick stainless steel gussets. The upper impact limiter is also a cask support member in the storage configuration, resting in the ISO cradle frame cross-member.

The cask lid is protected during impact by a 12-in. thick, balsa-filled lid impact limiter that covers and overlaps the cask lid and cavity flange. The balsa is enclosed within a 0.109-in. thick stainless steel container. A 0.125-in. sheet of asbestos is positioned between the balsa and the sheet material adjacent to the cask, and is contained internally. The impact limiter is attached to the cask lid by four 1-in. diameter bolts. There are elastomer O-rings in grooves under the heads of the 1-in. bolts and a neoprene gasket on the perimeter of the impact limiter. These impact limiter seals are for weather protection and do not provide a confinement function. Radiation exposure to the seals is not a failure concern. Removal of the lid impact limiter allows access to the cask lid.

Lifting devices for the NAC-I cask are designated as lifting trunnions and rotation trunnions. The lifting trunnions are two 8.625-in. diameter by 3-in. long trunnions located on the perimeter of the upper impact limiter. The cask is lifted by a special handling yoke attached to the two trunnions. The rotation trunnions are two 6.625-in. diameter by 3-in. long trunnions for

\textsuperscript{6} HeliCoil is a trademark of Cefilac, Societe Anonyme.
rotating the cask to and from the horizontal position in the ISO container. The lower trunnions are offset from the cask centerline so that when the cask is lowered into the ISO container, it will rotate to a horizontal position as the crane hook is lowered.

Transportation of the NAC-1 cask is in the normal storage configuration with the cask in a horizontal position, secured within the ISO container. Two structural cross-member sections serve as cradles for the cask within the ISO container. The lower (rotation) trunnions of the cask are captured by a notch and clamping plate on the aft cradle. The upper lifting trunnions are also captured by clamping plates, to hold the upper end of the cask as it rests on the impact limiter within a neoprene-lined forward cradle. The lid impact limiter is bolted to the cask lid. The lid impact limiter can be unbolted and rolled away from the cask on a track. The cask and ISO container are lifted by crane from the transport trailer and placed in the storage array on the assigned concrete pad.

**International Standards Organization Container Description**

The NAC-1 cask is transported and stored within a specially designed ISO shipping container. These containers are similar in design and appearance to SEALAND containers. The ISO container provides weather protection for the cask and does not perform a safety function. The ISO containers were fabricated in two heights, 6-ft. and 8-ft. high models. All containers are painted carbon steel construction. Figures D2-12 and D2-13 show the basic details of the ISO and internal structural support members.

The 8-ft. ISO container, fabricated by Evergreen Heavy Industrial Corporation, is nominally 8 ft. high x 8 ft. wide x 20 ft. long and has an empty weight of 8,624 lbs. The container has single full-width doors at each end and a removable roof. The frame of the container is structural steel channel construction, with a 0.50-in. carbon steel floor. The sides are fabricated of 0.063- and 0.079-in. corrugated carbon steel. The roof is fabricated of 0.063-in. carbon steel sheet metal supported by 1.575-in. x 1.575-in. angle iron on a 23.622-in. grid, and is slightly pitched to prevent ponding of precipitation. The doors and roof are provided with weather seals. Two of these containers will be used to house NAC-1 casks during fuel transport and storage.

The 6-ft. ISO container, fabricated by Adamson Containers Ltd., is 6 ft. high x 8 ft. wide x 20 ft. long and has an empty weight of 4,400 lbs. The container has two half-width doors at one end and a removable roof. Materials of construction and dimensions are similar to those used in fabricating the 8-ft. containers, except the roof material is 0.063-in. carbon steel sheet metal and the roof is flat. Four of these containers will be used to house NAC-1 casks during fuel transport and storage.

**D2.6.2.3.1 Primary Confinement.** Past inspection of the fuel assemblies and rods has shown no visible damage, cracks, or pin holes, and verification will be made during loose rod
packing, washing, drying, and loading into containers. This verification process meets the requirements of 10 CFR 72 for cladding inspection to ensure the integrity of the primary confinement. No additional testing is required.

**D2.6.2.3.2 Secondary Confinement.** With fuel cladding providing the primary confinement, the seal-welded canisters provide the secondary confinement boundary and eliminate the need to rely on the storage cask for confinement. The welded canister also satisfies the fuel retrieval requirement of 10 CFR 72. Prior to seal welding, the inner canisters will be backfilled with helium (at approximately 1 atm) for leak detection purposes. The presence of helium is not required for heat transfer purposes, nor to maintain fuel integrity. The canister is designed to eventually be transloaded into a transportation system using the DOE standardized SNF canister (18 in.). Each spent fuel canister will be contained within a NAC-1 cask for shipment and storage at the Hanford Site.

**D2.7 SAFETY SUPPORT SYSTEMS**

This section identifies and describes the principal systems that perform safety support functions.

**D2.7.1 Criticality Prevention**

**D2.7.1.1 Fast Flux Test Facility Fuel.** A criticality safety evaluation was performed for storage of the FFTF SNF in the 400 Area ISA (WHC-SD-FF-CSER-004) and provides the bases for satisfying the criticality prevention criteria of double contingency to $K_{\text{eff}} < 0.95$. Since the configuration will not change for the 200 Area ISA, the calculations and evaluations performed for the 400 Area ISA are valid. In particular, the CCC was analyzed with respect to its structural and confinement effectiveness as a canning structure for FFTF fuel. The results for the CCC demonstrate that the CCC retains its confinement and structural integrity under all normal and accident conditions where all loads were found to be within the Code Class limits, g allowables, and stress allowables for the CCC.

**D2.7.1.2 NRF TRIGA Fuel.** There are no components performing a criticality prevention function requiring safety classification (Chapter D6.0). The criticality safety evaluation was performed for the storage of NRF TRIGA SNF in the 400 Area ISA and provides the bases for satisfying the criticality prevention criteria including double contingency and $K_{\text{eff}} < 0.95$. The TRIGA fuel configuration at the 200 Area ISA is identical to the 400 Area configuration.

**D2.7.1.3 Commercial Light Water Reactor Fuel.** Separate criticality safety evaluation reports were prepared to address interim storage of PWR fuel assemblies and loose fuel rods including BWR fuel in the NAC-1 casks in the ISA. These reports and Chapter D6.0 address specific
criticality aspects of LWR fuel storage in the ISA as individual casks and in the storage array. The LWR canister is required for criticality geometry control and is designated Safety Class.

The canisters for the PWR fuel assemblies, and the canister internals, will provide the safety-class support to retain the fuel material within the square cross-sectional geometry of the fuel assembly. For the BWR and PWR loose rods, the rod container and/or canister and internals will provide the safety-class limit for the fuel pitch of the bundle of rods to a corresponding diameter of 10.9 in.

D2.7.2 Fire Protection

The ISA is constructed of non-combustible materials and there are no fire hazards inherent to the ISA (SNF-4932). Administrative controls ensure that the area within the fence is kept free of debris and vegetation. The fence, provided for access control and radiological exposure control purposes, will also keep transient debris from entering the ISA. The only credible fire hazard associated with the ISA is the equipment used to transfer and handle the storage systems. Accidents associated with the ISA have been determined to be bounded by the transportation fire scenario of 10 CFR 71.73(c)(3). Specific accidents that are bounded include the transporter tractor vehicle, mobile crane fire, and runaway fuel truck. Therefore, the maximum credible fire for the ISA is bounded by the transportation fire scenario.

D2.7.2.1 Fast Flux Test Facility Fuel. The ISC is constructed of nonflammable materials. The design basis storage fire for the ISC is bounded by the design basis transportation fire defined in 10 CFR 71, as discussed in the 200 Area ISA fire hazards analysis (SNF-4932).

D2.7.2.2 NRF TRIGA Fuel. The TRIGA cask is constructed of nonflammable materials. The survival of the TRIGA cask is demonstrated by the analysis of the design basis fire. Since the NRF TRIGA cask is removed from the impact limiters that were analyzed as part of the transportation fire in the SARP (WHC-SD-TP-SARP-008), a supplemental fire analysis was performed for the storage configuration inside the Rad-Vault.

The DOT-6M container is constructed of nonflammable and fire retardant materials.

D2.7.2.3 Commercial Light Water Reactor Fuel. The NAC-1 and the ISO are constructed of nonflammable materials. The survival of the NAC-1 is demonstrated by the analysis of the design basis fire.
D2.7.3 Cask Instrumentation

Continuous monitoring of parameters of the dry storage systems (e.g., interseal pressure and temperature) are not provided. While such monitoring is recommended for mechanical closures (NUREG-1536, page 7-3), all three types of fuel—FFTF, TRIGA, and LWR—have demonstrated intact fuel cladding prior to placement into storage, and the storage system confinement boundaries are tested to a leak tight condition, where leaktight is as defined by ANSI N14.5. The FFTF and TRIGA fuels are protected by multiple confinement barriers and have smaller releasable gas and volatile radionuclide source terms compared to commercial LWR fuel dry storage systems. The LWR fuel stored in the NAC-1 casks is confined in a seal welded container, which is also tested to leaktight conditions. Large margins in the thermal design of the storage systems precludes the need for temperature monitoring to ensure fuel integrity. The dry storage systems are described below.

Surveillance of the ISA is required to assure safe storage (NUREG-1536, page 7-3), when cask parameter monitoring is not used. This surveillance assists in the early identification of conditions that could lead to unsafe storage. The surveillance requirement is met by a combination of closed circuit television (CCTV) monitoring and/or frequent (daily) walk through (or walk by) inspections for fence security, fire watch, roving patrol, or other periodic checks. Surveillance procedures and closed circuit television design will be implemented prior to ISA operation. CCTV monitoring will be in a continuously manned location (i.e., CSB control room or patrol headquarters).

D2.7.3.1 Fast Flux Test Facility Fuel. Only intact FFTF fuel is contained within the FFTF ISC, which is a stainless steel and concrete cask, with the confinement boundary provided by the stainless steel shell, closure lid, and double metallic O-ring seals. The confinement boundary is designed to the ASME Code, Section III, Subsection NC. The double metallic O-ring seals are tested to leaktight conditions, and then the penetration port between the O-ring seals is closed by welding the penetration port covers in place. An additional cover plate may be seal welded over the bolted closure plug after receipt at the 200 Area ISA to enhance the long-term storage configuration.

Because none of the DBAs can result in a breach of the ISC confinement boundary, and normal storage environmental conditions are well within the design parameters of the metallic Helicoflex double seals, continuous pressure monitoring is not provided for the interseal space.

The FFTF ISC uses air channels to provide for cooling air flow along the walls of the vessel that provide confinement. Historically, the NRC has required the monitoring of outlet temperatures of casks to demonstrate that the thermal performance of the cask is not degrading. Thermal analysis (WHC-S-4100) has determined that the cask thermal performance is acceptable with the absence of cooling air flow. Thus, no loss of confinement events occur due to high
temperature. No monitoring of the outlet air temperature of the FFTF ISC is required or provided.

The FFTF ISC design incorporates provisions for continuous monitoring of cask parameters, if needed for future use. Conduit runs for wiring have also been provided in the FFTF storage pad of the 200 Area ISA.

**D2.7.3.2 NRF TRIGA Fuel.**

**NRF TRIGA Cask**

TRIGA fuel that is considered to be intact is loaded into NRF TRIGA casks. The TRIGA casks are closed after loading and leak tested to leaktight conditions by testing between a metallic closure seal and a Viton O-ring.

This design does not meet the requirement for double metallic O-rings (NUREG-1536, page 7-2), but this fuel has a very small source term for gas and volatile radionuclides. Since the potential releasable source term is small and there are no DBA mechanisms for seal failure, monitoring of the confinement boundary does not seem justified and is not provided.

The TRIGA cask incorporates passive heat rejection from the body of the cask and does not require thermal performance monitoring.

The TRIGA cask design incorporates provisions for continuous monitoring of cask parameters, if needed for future use. Conduit runs and wiring can also be provided to the Rad-Vault gravel area, if necessary.

**DOT-6M Container**

TRIGA FFNCRs are loaded into a DOT specification 2R container. The 2R container incorporates a leak test port and double O-rings (one metallic and one elastomer). When closed, the 2R is tested to demonstrate a leak tight seal.

The design of the 2R container incorporates a metal seal in the outer groove of the container and an elastomer Viton O-ring in the inner groove. If the Viton O-ring fails, the interscell test port becomes part of the confinement boundary. To preclude possible leakage from the test port, it is welded closed after receipt at the 200 Area ISA. Radiation exposure to the seals is not a failure concern.

Since the potential releasable source term is very small and there are no DBA mechanisms for seal failure, monitoring of the confinement boundary does not seem justified and is not provided.
The 6M drum incorporates passive heat rejection from the body of the drum and does not require thermal performance monitoring.

There are no design features that can be used for parameter monitoring of the 2R/6M drum configuration. Consequently, parameter monitoring cannot be implemented for this configuration.

**D2.7.3.3 Commercial Light Water Reactor Fuel.** The LWR fuel stored in the NAC-I casks is considered to be intact. The fuel is confined in the LWR canister, which is closed by welding. After it is welded closed, the canister is tested to leaktight conditions, as defined by ANSI N14.5. This configuration provides two confinement boundaries and is considered to meet the requirement of assured confinement. In accordance with NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems* (page 7-4), no confinement monitoring will be required. After loading the LWR canister into the cask, the cask will be closed using the containment boundary neoprene O-rings for transportation, but these O-rings are not required to maintain integrity for extended storage.

The NAC-I cask incorporates passive heat rejection from the body of the cask and does not require thermal performance monitoring. The decay heat of the stored LWR fuel is significantly less than the design basis heat load of the cask. The NAC-1 cask is designed for heat loads up to 750 W (NAC 1990).

Since the LWR canister is a welded closure, no confinement monitoring is required. The NAC-1 cask and LWR canister designs do not incorporate provisions for continuous monitoring of any cask parameters.

**D2.7.4 Radiation Monitoring**

10 CFR 72, Sections 122(1) and 126(c)(2), require the ability to monitor radiation levels. At the ISA, since the radiation doses are not anticipated to change after cask placement, the most appropriate means of doing this is dosimetry, alarming supplemental dosimetry, and periodic measurements by radiological control technicians. An area radiation monitor is not provided. Thermoluminescent dosimeter monitoring on the 200 ISA perimeter fence will demonstrate compliance with radiation dose limits imposed by 10 CFR 72 and monitored in accordance with 10 CFR 835.
D2.7.5 Cranes

The cranes to be used include the following:

- 150-ton Manitowoc 4000W crawler
- 250-ton Manitowoc 4600 crawler
- Hydro-Crane for TRIGA.

The above cranes have been analyzed for the crane fall accident in Section D3.4.2. A Technical Safety Requirement precludes use of other cranes in the ISA unless analyzed and approved.

D2.7.6 Lifting Equipment

All hoisting and rigging evolutions will be conducted in accordance with DOE/RL-92-36. All lifts of SNF or casks containing SNF will be critical lifts.

D2.7.6.1 Fast Flux Test Facility Fuel. The lifting equipment interfaces with the three removable ISC lifting lugs attached to the 150-ton crane (Figure D2-2). The three ISC lifting lugs are equally spaced on a 62.0-in. diameter circle on the top of the ISC. The spacing aligns directly with the existing spreader assembly (drawing H4-65153-A1). Three existing tension-ties (drawing H4-65153-A2) are used to connect the spreader assembly to the 1.75-in. thick lifting lugs. The fully loaded ISC weighing 114,200 lbs. uses 86% of the safe working load of the three tension-tie assemblies. Lifting will be covered by an approved procedure.

D2.7.6.2 NRF TRIGA Fuel. Lifting of the TRIGA cask is performed with two swivel hoist rings that thread into the cask cover. These rings are connected with two slings to a lifting beam, which connects to the crane hook with single slings and a load cell. The loaded TRIGA cask weighs approximately 1,900 lbs. (see Figure D2-16). Lifting will be covered by an approved procedure.

The DOT-6M is lifted with a commercial drum lifter rated at 1,000 lbs. capacity, which connects to the crane hook using single slings and a load cell. The DOT-6M weighs approximately 650 lbs. (see Figure D2-17). Lifting will be covered by an approved procedure.

The Rad-Vault lid is lifted with four slings connected to a lifting beam, which then has two slings to the crane hook with a minimum sling angle of 60 degrees. The Rad-Vault lid weighs approximately 20,000 lbs. (see Figure D2-18). Lifting will be covered by an approved procedure.
**D2.7.6.3 Commercial Light Water Reactor Fuel.** The lifting equipment interfaces with the four ISO top corner lifting lugs, the hydraulic ISO lifting fixture, and with the 250-ton crane. The fully loaded ISO weighs a maximum of 56,000 lbs. Lifting will be covered by an approved procedure.

**D2.8 UTILITY DISTRIBUTION SYSTEMS**

**D2.8.1 Interim Storage Area Power Supply System**

The ISA has power at the fence for the sodium lighting system. The lighting provides a security function for the required drive-by inspections.

The conduit embedded in the concrete pads is PVC GRC meeting the NEMA Standard RN-1. The conduits are terminated with a rigid steel coupling and pipe plug, installed flush with the top of the concrete. Conduit ends terminating in vaults or handholds are terminated with ground bushings. Ground bushings are connected in series with a minimum of #12 AWG bare copper wire to six ground conduits of 0.75 in. diameter.

The ISC pads have conduits for power or signals. There are five handholds located along the outer longer edges of each ISC pad. The first being ~2.5 ft. from the north edge, and the next four being spaced at approximately 30 ft. to each successive one. After the fourth handhold, to the south, the PVC GRC drops from a 4-in. to 2-in. diameter, and is a single run to the fifth handhold. Signal conduit runs from the handhold out under the concrete pad and terminates to the six successive storage locations. This conduit is buried at a depth of 24 in. below grade. The power and signal conduits become coincidental at 24 ft. from the northern-most outer edge of each ISC concrete pad.

The NAC-1/ISO pad has conduits for power and signals. The single handhold for the NAC-1 ISO seven conduits of 0.75-in. PVC GRC run from the handhold, and one of the seven is terminated at each of the seven proposed storage locations. They are located 1-ft. north and 3-ft. west of the northeast corner of each storage container. This conduit is buried at a depth of 24 in. below grade. The power and signal conduits become coincidental, as the pad is centrally located along the northern 28-ft. edge at 2.50 ft. from the pad, 35 ft. from the northern-most outer edge of each NAC-1/ISO concrete pad.
D2.8.2 Interim Storage Area Piped Utility System

One fire hydrant (1,000 gal/min) is located approximately 152 ft. from the southwest corner of NAC-1/ISO pad, just outside the fence line (Drawing H-2-829294). There is also an 8-in. fire main connection at the proposed ISA storage building.

D2.8.3 Interim Storage Area Communications System

Currently, no monitoring system is identified for use at the ISA. Conduit is provided for flexibility and future use. Communication will be provided via portable handheld equipment.

The conduit embedded in the concrete pads is PVC GRC meeting the NEMA Standard RN-1. The conduits are terminated with a rigid steel coupling and pipe plug, installed flush with the top of the concrete. Conduit ends terminating in vaults or handholds are terminated with ground bushings. Ground bushings are connected in series with a minimum of #12 AWG bare copper wire to six ground conduits of a 0.75-in. diameter.

The ISC pads have 4-in. and 2-in. conduits for signal wires. There are five handholds located along the outer longer edges of each ISC pad. The first being ~2.5 ft. from the north edge, and the next four being spaced at approximately 30 ft. to each successive one. After the fourth handhold, to the south, the PVC GRC drops from a 4-in. to 2-in. diameter and is a single run to the fifth handhold. Signal conduit runs from the handhold out under the concrete pad and terminates to the six successive storage locations. This conduit is buried at a depth of 24 in. below grade. The power and signal conduits become coincidental at 24 ft. from the northernmost outer edge of each ISC concrete pad.

The NAC-1/ISO pad has 2-in. conduits for signal wires. The single handhold for the NAC-1/ISO pad is centrally located along the northern 28-ft. edge, 2.50 ft. from the pad. Seven conduits of 0.75-in. PVC GRC run from the handhold, and one of the seven is terminated at each of the seven proposed storage locations. They are located 1-ft. north and 3-ft. west of the northeast corner of each storage container. This conduit is buried at a depth of 24 in. below grade. The power and signal conduits become coincidental at 35 ft. from the northern most outer edge of each NAC-1/ISO concrete pad.
D2.9 AUXILIARY SYSTEMS AND SUPPORT FACILITIES

D2.9.1 Lightning Protection

D2.9.1.1 Fast Flux Test Facility Fuel. A lightning analysis has been performed for the FFTF ISC (SNF-4791), demonstrating that lightning strikes will not credibly lead to the release of radioactive materials from the ISC. The analysis shows that effects of a lightning strike to the ISC cask are minimal. A direct strike on the cask would not produce significant temperature rise or damage to the cask contents. Subsequent to a likely voltage breakdown in the top concrete layer during a direct, worst-case 218,000 amp lightning strike to the cask, current flow during the discharge would pass along the concrete reinforcing rod structure and from there to ground. Current flow through the internal cask materials inside of the inner shield would be negligible due to the electrical skin effect. Lightning cannot lead to the release of radioactive material; therefore, no lightning protection is required.

D2.9.1.2 NRF TRIGA Fuel. A lightning analysis was performed (SNF-4793, Section 3.0) that evaluated the effects of a lightning strike on a Chem-Nuclear Systems, Inc. Type 14-215 Rad-Vault storage container. The container consists of a concrete structure with steel reinforcement. The analysis assumed that the cask was struck with the full energy of the maximum credible lightning strike. The analysis shows that effects of a lightning strike to the Rad-Vault cask are minimal. A direct strike on the cask would not produce a significant temperature rise or damage to the cask contents. Subsequent to likely voltage breakdown in the top concrete layer and during the event of a direct worst-case 218,000 ampere lightning strike to the cask, current flow during the discharge would pass along the concrete reinforcing rod structure and from there to ground. Current flow through the internal regions of the Rad-Vault wall would be negligible due to the electrical skin effect. Spalling of concrete at both the top and bottom areas of the cask could occur at the regions where the current penetrates the concrete and enters or exits the steel reinforcing wires. Arc discharge would likely occur in the gasket area, possibly exposing interior gases to an ignition source. No flammable gases are anticipated inside the vented Rad-Vault containing leaktight containers. Lightning cannot lead to the release of radioactive material; therefore, no lightning protection is required.

D2.9.1.3 Commercial Light Water Reactor Fuel. An analysis was performed that evaluated damage as a result of a lightning strike to the cask/container system (SNF-4795). The analysis concludes that even in the most severe case where the cask is not protected by the surrounding ISO shipping container, a direct lightning strike would not produce significant temperature rise or damage to the cask surface materials, fuel contents, or confinement function.

Current flow during the discharge of a lightning strike would pass along the stainless steel surface of the cask and from there to the ground. Adequate electrical bonding between the cask and storage container is provided by the cask rotation trunnion clamps. Current flow
through the interior of the cask would be much less than the surface due to the electrical "skin
effect." In the event that a maximum current level strike occurred on the cask top end, current
flow would spread along the surface on its path to ground. Because of the metal-to-metal contact
near the outer surface and far from the seals, negligible current would pass through the
Helicoflex metallic seals. Maximum temperature rise at the surface of the cask during the
lightning strike is estimated to be 28 °F. With negligible current flow through the metallic seals,
the temperature rise in the seals would be even lower than the cask surface temperature. Since
the metallic seals are designed for a maximum temperature of 700 °F, the temperature rise
expected during a lightning strike would not result in damage to the metallic seal, and
confinement of the fuel would be preserved.

The lightning analysis does not take credit for the cask being stored in the ISO shipping
container. The ISO shipping container provides additional lightning protection for the cask, as it
will dissipate current in a similar manner as the cask surface during a lightning strike, preventing
any flashover of electrical current to the cask. No lightning protection is required.

D2.10 REFERENCES

Regulations.

10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and


10 CFR 830, "Nuclear Safety Management," Section 830.120, "Quality Assurance


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Institute, Farmington Hills, Michigan.

ACI-224, 1980, Control of Cracking, American Concrete Institute, Farmington Hills, Michigan.

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Michigan.

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NRMCA, 1988, *Certification of Ready Mixed Concrete Production Facilities*, National Ready Mix Concrete Association, Silver Spring, Maryland.

Annex D – 200 Area Interim Storage Area


UCRL-15910, 1988, Design and Evaluation Guidelines for Department of Energy Facilities Subjected to Natural Phenomena Hazards, Lawrence Livermore National Laboratory, Livermore, California.


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Figure D2-1. 200 Area Interim Storage Area.
Figure D2-2. Interim Storage Cask with Lift Fixture Attached to Mobile Crane.
Figure D2-3. Rad-Vault.
Figure D2-4. NRF TRIGA Cask - Arrangement.
Figure D2-5. NRF TRIGA Cask - Outer Vessel.
Figure D2-6. DOT-6M Container.
Figure D2-7. NRF TRIGA Cask - Fuel Configuration After Third Shipment.
Figure D2-8. Interim Storage Cask.
Figure D2-9. Core Component Container.
Figure D2-10. NAC-1 Cask.
Figure D2-11. NAC-1 Cask/Light Water Reactor Canister System.
Figure D2-12. International Standards Organization Shipping Container - External Structure.

SHIPPING CONTAINER (8-ft Evergreen)

External Structure

Top, 0.063 Steel
(0.055 Adamson 6ft)

Sides 0.063-0.079
Corrugated Steel
(0.079 Adamson 6ft)
Figure D2-13. International Standards Organization Shipping Container - Internal Structure.

SHIPPING CONTAINER (8-ft Evergreen)(*6-ft Adamson)

Major Internal Members

- Removable Header
- 5.1" x 4.8" Box Section (2 places)
- 6.2" x 4.8" Box Section (4 places)
- 9.8" Deep Channel
- 3.5" Flange (4 places)
- 5.1" Deep Channel
- 6.3" Top Flange
- 4.8" Bottom Flange (2 places)
- Beam Spacing variable 6" to 12"

Floor Beams

- 5.1" Deep Channel
- 1.8" Flange

Dimensions:
- Height: 66.0" (Total height)
- Height: 30.0" (Upper height)
- Height: 11.0" (Lower height)
Figure D2-14. Light Water Reactor Fuel Canister Assembly.
Figure D2-15. Loose Pin Container.
Figure D2-16. NRF TRIGA Cask Lift Equipment.

- **CRANE HOOK**
  - MIN. SWL 3,000 LBS.

- **SLING**
  - MIN. SWL 3,000 LBS.
  - SHACKLE
  - MIN SWL 3000 LBS.

- **LIFTING BEAM**
  - DWG# H-2-821475
  - SHACKLE (2 PLACES)
  - MIN SWL 1000 LBS.

- **SLING (2 PLACES)**
  - MIN. SWL 1000 LBS.

- **SWIVEL HOIST**
  - RINGS (2 PLACES)
  - MIN. SWL 1000 LBS.
  - 3/8-16-UNC X 54 LENGTH

- **NRF TRIGA CASK LIFTING BEAM ASSEMBLY**
  - MIN. SWL 2,200 LBS.

- **NRF TRIGA CASK**
  - 1,900 LBS.
Figure D2-17. DOT-6M Lift Equipment.
Figure D2-18. Rad-Vault Lid Lift Equipment.

- **Crane Hook**
  - Min SWL 20 Ton Cap.

- **Shackle**
  - Min SWL 20,000 Lbs
  - Min Sling Angle 60°

- **2 Way Sling**
  - Min SWL 24,000 Lbs
  - (Per Leg 12,000 Lbs)
  - Min Length 6 Ft

- **Lifting Shackle Beam Assembly**
  - Min 40 Ton
  - #10389-3-4/3/92

- **Lifting Trunnions**
  - Sling (4 Places)
  - Min SWL 10,000 Lbs
  - (Each Sling)
  - Min Length 6 Ft

- **Top - Rad Vault**
  - 20,000 Lbs

- **Rad Vault**
CHAPTER D3.0

HAZARD AND ACCIDENT ANALYSES
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D3.0 HAZARD AND ACCIDENT ANALYSES

D3.1 INTRODUCTION

This chapter presents a summary of the key methodology, assumptions, and results of the final safety hazard analysis and design basis accident (DBA) analyses performed for the final design and operation of the 200 Area Interim Storage Area (ISA). These analyses form a safety basis for the final safety analysis report (FSAR) and present a comprehensive evaluation of the ISA handling- and storage-related activities and the natural phenomena and external hazards that can affect the public, workers, and environment. Single and multiple initiating events from equipment and human-error failures in the facility, and human and natural events outside of the facility, have been considered.

The contents of this chapter are as follows:

- The requirements for establishing the safety basis for the ISA are listed in Section D3.2. The requirements listed consist of U.S. Department of Energy (DOE) Orders and standards and applicable U.S. Nuclear Regulatory Commission (NRC) rules and guidance.

- The ISA hazard analysis is summarized in Section D3.3. The complete hazard analysis is contained in SNF-4820, 200 Area Interim Storage Area Final Hazard Analysis. The analysis identifies hazard sources, hazardous conditions, potential accident scenarios and their initiators, and preliminary assessments of event frequencies and consequences. Hazards are identified by form and location and represent a complete spectrum of events that could occur throughout the facility. An initial set of safety features that would serve to prevent or mitigate the postulated accident scenarios is identified in the hazard analysis, with a final set of safety features identified in the accident analyses in Section D3.4.2 and in Chapters D4.0 and D5.0. Hazard analysis methodology and evaluation criteria are discussed in further detail in Section D3.3.

- The final facility hazard classification, determined in accordance with DOE-STD-1027-92, Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports, is addressed in Section D3.3.2.2. The ISA has been assigned a final designation of a Hazard Category 2 facility based on material at risk.

- Section D3.3.2.3 contains discussions of defense in depth, worker safety, and environmental protection, including a detailed tabulation of engineered and administrative features that have been identified as providing for worker safety.
High-risk events that pose a challenge to off-site release limits and on-site radiological dose evaluation guidelines have been selected from the hazard analysis for further detailed quantitative evaluation as DBAs. These DBAs are presented in Section D3.4.2.

Three receptor locations were used for the DBA analysis:

- Hanford Site boundary (17,390 m east of the ISA) — defined release limits; used for calculation of off-site doses and selection of safety-class features.
- Collocated worker (100 m from the ISA) — defined risk evaluation guidelines; used for calculation of on-site doses and selection of safety-significant features.
- Highway 240 (on-site, approximately 9,280 m west of the ISA) — no defined evaluation guideline; doses calculated for information purposes only (Scott 1995).

Safety-class and safety-significant features have been selected for each of the analyzed DBAs from the candidate structures, systems, and components (SSCs) identified in the hazard analysis. When required by the Spent Nuclear Fuel (SNF) Project’s commitments to meet equivalent NRC requirements, candidate features have been identified as "important to safety" and designed, engineered, and procured consistent with safety-class or safety-significant classification requirements. Safety-class and safety-significant features are presented first in Section D3.4.2, with the discussion of each accident, and described in more detail in Chapters D4.0 and D5.0.

Each of the DBAs that have been quantitatively analyzed represents a bounding case for a category or type of hazards and accidents. An in-depth review was performed of all significant accidents identified by the hazard analysis. The table and text that accompany each DBA in Section D3.4.2 include the preventive and mitigative features and the associated Technical Safety Requirements (TSRs) for the bounding case presented and for all other events binned within the accident category. Defense-in-depth features are also identified in these tables.

Chapter D3.0 interfaces with several other SNF Project safety documents. The planned scope and content of various other SNF Project safety reports, TSRs, and supporting safety documents are defined in Administrative Procedure MS-1-039, Spent Nuclear Fuel Project ISMS Description. The hazards associated with transport and accident scenarios that are postulated during shipment are analyzed in the following safety analysis report for packaging (SARP) documents:

- WHC-SD-TP-SARP-010, 1996, Safety Analysis Report for Packaging (Onsite) Interim Storage Cask

D3.2 REQUIREMENTS

Chapter 3.0 of the SNF Project FSAR lists the design codes, standards, regulations, and DOE Orders that contain requirements and guidance for establishing the safety basis for the SNF Project. Specific codes, standards, and requirements applicable to the 200 Area ISA are defined in the SNF Project Standards/Requirements Identification Document (HNF-SD-SNF-RD-001). Only the requirements that are specific for the ISA and that pertain to the safety analysis are provided here.

- **DOE Order 5480.23, Nuclear Safety Analysis Reports**, in conjunction with its Attachment 1, "Interim Guidance for DOE Order 5480.23," sets the requirements for analysis. This chapter complies with the requirements by documenting the performance of hazard and accident analyses. The analyses were performed to the guidance of HNF-PRO-704, Hazard and Accident Analysis Process, which ensures that the hazards and accident analyses comply with DOE Order 5480.23. The methodology, assumptions, and criteria used to identify facility hazards, hazard rankings, candidate accidents, DBAs, preventive and mitigative features and controls, and the classification of these features (along with the definition of safety functions, performance criteria, and applicability) are described in Chapter 3.0 of the SNF Project FSAR and documented in this chapter.

- **DOE Order 5480.22, Technical Safety Requirements.** This Order sets the requirements for developing and preparing a TSR document. This chapter complies with the requirements by documenting the performance of hazard and accident analyses in accordance with HNF-PRO-704 and DOE Order 5480.23. The results of the analyses were used to identify specific SSC safety functions, performance requirements for the SSCs, and the times for application of the safety functions.

- **DOE Order 6430.1A, General Design Criteria.** This Order provides requirements for the identification of safety-class items. The analyses documented in this chapter used the SSC classification requirements of DOE Order 6430.1A in the identification of safety-class SSCs. Compliance with DOE Order 6430.1A is demonstrated in SNF-5139, **DOE 6430.1A Compliance Analysis for the 200 Area Interim Storage Area.**

The following standards are used for content and guidance:

- **DOE-STD-1027-92, Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports.** This
standard provides guidance for the preparation and review of hazard categorization and accident analysis techniques, as required by DOE Order 5480.23. Of particular importance to this chapter is the guidance pertaining to the hazard categorization methodology and the accident analysis techniques that are appropriate for the graded approach required by DOE Order 5480.23. Section D3.3.2.2 documents the final facility categorization of the ISA, determined in accordance with DOE-STD-1027-92.

- **DOE-STD-3009-94, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports.** This standard supplements DOE Order 5480.23 by providing guidance specific to nonreactor nuclear facilities. In this regard, the standard provides detailed information on the performance of accident analyses for Hazard Category 2 and 3 facilities. The standard also provides guidance for establishing defense-in-depth measures and addresses safety-significant SSCs. This chapter has been organized and prepared consistent with the specifications of DOE-STD-3009-94.

In Letter 95-SFD-167, *Implementation of K Basins Spent Nuclear Fuel Project (SNFP) Regulatory Policy* (Sellers 1995), DOE established the requirement for new SNF Project facilities to achieve "nuclear safety equivalency" to comparable NRC-licensed facilities. The SNF Project identified the NRC requirements that were to be met, in addition to existing and applicable DOE requirements, to establish nuclear safety equivalency. These NRC requirements, and the process used to identify them, are documented in HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements*, and in WHC-SD-SNF-DB-009, *Canister Storage Building Natural Phenomena Hazards*, Appendix C, "200 East Area Interim Storage Area Natural Phenomena Hazards."

The requirements for achieving NRC equivalency include the following:

- **Title 10, Code of Federal Regulations, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR 72).** This rule is used for licensing independent spent fuel storage installations. Section 72.122, "Overall Requirements," requires that the design bases for SSCs important to safety reflect appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. Section 72.24, "Content of Application: Technical Information," provides requirements in Paragraph 72.24(m) for the analysis of accidents and natural phenomena events that could result in a dose at the controlled area boundary.

- **NRC Regulatory Guide 3.48, Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage).** This guide establishes the format and content for safety analysis reports for license applications of fuel storage facilities.
Important-to-safety SSCs have been identified in accordance with 10 CFR 72.3. Once SSCs have been identified as having a function meeting the definition of important to safety, the requirements for SSCs important to safety specified in 10 CFR 72 are imposed. A graded approach is applied to an SSC important to safety by using the guidance provided in NUREG/CR-6407, Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety, as follows:

- **Category A — Critical to Safe Operation**

  SSCs in this category include those whose failure or malfunction could directly result in a condition adverse to public health and safety. Important-to-safety SSCs in this category are classified as Safety Class, as defined in DOE Order 6430.1A with the additional requirements therein.

- **Category B — Major Impact on Safety**

  SSCs in this category include those whose failure or malfunction could result in a condition adversely affecting collocated worker health and safety. Note that from the definition of Category C, Category B is understood to include events that could significantly damage the storage containers without severe impact to public health and safety. SSCs in this category are classified as Safety Significant.

- **Category C — Minor Impact on Safety**

  SSCs in this category include those whose failure or malfunction would not significantly reduce containment and would not be likely to create a situation adversely affecting public or collocated workers' health and safety. SSCs in this category are classified as General Service.

While these documents have particular significance to this chapter, they do not, by themselves, establish requirements for the ISA or the Chapter D3.0 accident analyses. Requirements identified in HNF-SD-SNF-DB-003 and WHC-SD-SNF-DB-009 address NRC equivalency. It is noted that the SNF storage casks are "important to safety" based on the 10 CFR 72.3 definitions. The SNF Project has committed to comply with select NRC requirements. Additional requirements include safety documentation for (1) determining important-to-safety classifications for preventive and mitigative features; (2) identifying and resolving worker safety issues in DOE Order 6430.1A, DOE Order 5480.23, and DOE-STD-3009-94; and (3) evaluating nearby activities. The determination of important to safety is directly incorporated into the identified preventive or mitigative safety function features at the end of each DBA scenario in Section D3.4.2, and external human-generated threats are discussed in Section D3.3.2.3. Requirements of WHC-SD-SNF-DB-009 are incorporated into the natural phenomena hazard (NPH) design criteria and are considered in the hazard evaluation.
Letter 97-SFD-172, *Risk Evaluation Guidelines (REGs) to Ensure Inherently Safer Designs* (Sellers 1997), provides another SNF Project requirement significant in the development of this chapter (e.g., safety-significant SSCs).

**D3.3 HAZARD ANALYSIS**

The hazard identification and evaluation process provides a thorough, predominantly qualitative evaluation of the spectrum of risks to the public, workers, and the environment caused by accidents that involve the hazards identified in the hazard analysis. This process is further described in Section 3.3 of the SNF Project FSAR. An initial hazard analysis (HNF-2015) was prepared to support HNF-2524, *200 East Area Interim Storage Area Preliminary Safety Evaluation Report*, and to integrate all previous ISA hazard analyses. A final hazard analysis has been performed to support the accident analysis and is documented in SNF-4820. The final hazard analysis systematically reviewed the final ISA design, as described in Chapter D2.0, associated supporting design documentation, and design references to identify any additional hazardous materials or energy sources that have the potential to initiate an accident that could require further review or analysis. This process resulted in the selection of seven DBAs for more comprehensive analysis in Section D3.4.2.

**D3.3.1 Methodology**

The methodology used to identify and evaluate the SNF Project facility hazards is described in detail in Section 3.3 of the SNF Project FSAR. This section discusses the areas of this methodology that are specific to the ISA. The hazard evaluation process identified hazardous conditions, determined causes and preventive and mitigative features, and qualitatively estimated the consequences and frequencies of occurrence. The results of the application of this methodology to the ISA are presented in Section D3.3.2. The hazard analysis was performed in accordance with DOE-STD-3009-94 and implements the requirements of DOE Order 5480.23.

**D3.3.1.1 Hazard Identification.** No adverse consequences for the public or workers, or that cause contamination of the environment, are expected under normal operations or storage. The hazard analysis focused on abnormal and accident conditions, as described in SNF-4820. The hazards associated with transport and the accident scenarios that are postulated during shipment are analyzed in the SARPs.

The ISA cask handling and storage activities that can take place within the boundary of the storage area were identified, and the hazards were identified by form (e.g., electrical, thermal, friction), type (e.g., motors, power tools, wiring), and location. A standardized hazardous material/energy source checklist (see Section 3.3.1.1, Table 3-2, of the SNF Project FSAR) was used to group the potentially hazardous materials and energy sources. The methodology of the
hazards identification process is described further in the Section 3.3.1.1 of the SNF Project FSAR.

**D3.3.1.2 Hazard Evaluation.** As described in Section D3.3.1.1, hazards associated with abnormal and accident conditions at the ISA were identified. The hazards identified were evaluated to determine the causes of the hazard, potential accidents that could result from the presence of each hazard, and consequences to the public off-site, the collocated and facility workers on-site, the environment, or the ISA. Safety features, segregated into preventive and mitigative features, were identified for each hazard based on the ability of the feature to prevent or mitigate the consequences. Qualitative estimates of the frequency and consequences of the hazardous condition also were assigned (see Section 3.3.1.2 of the SNF Project FSAR for the criteria used in assigning the consequence and frequency categories).

**D3.3.2 Hazard Analysis Results**

The hazard analysis process is described in SNF-4820 and summarized in Section D3.3.1 and in Section 3.3.1 of the SNF Project FSAR. The products of that process, in the order of progression, are as follows:

1. Develop a series of checklist-style tables describing hazardous materials and energy sources, organized by fuel type. These tables were used to develop hazard analysis accident scenarios.

2. Develop a series of tables describing the standard industrial hazards considered, organized by fuel type. These events were judged to have no contribution to uncontrolled radiological and/or hazardous materials releases and were not considered in the selection of DBAs, safety-class or safety-significant features, or TSRs. They were among the hazards considered and, therefore, are included in this analysis for completeness.

3. Develop a series of tables describing potential hazard scenarios. These tables include hazardous energy sources and materials, hazardous conditions, causes and initiators, potential accidents, qualitative determinations of event frequencies and consequences, safety features for prevention and/or mitigation of the consequences, and defense-in-depth or worker safety features.

4. Compile a table assigning risk bins to causes associated with significant consequences to off-site and on-site receptors. Consistent with DOE-STD-3009-94, the events located in risk bins representing "situations of concern" or "situations of major concern" were evaluated as candidate DBAs.
5. Prepare a final list of candidate DBAs sorted by risk ranking and energy change or release. This list formed the basis for selection of the DBAs presented in Section D3.4.2.

DBA selection is addressed in Section D3.3.2.3.5 and its accompanying Table D3-5. In terms of the risk binning process, the accidents chosen from the hazard analysis for further analysis as DBAs were all events identified in consequence categories S3 and S2. These categories indicate significant effects to off-site and on-site receptors.

The final hazard analysis has been reconciled with the up-to-date ISA facility design described in Chapters D2.0, D4.0, and D5.0.

D3.3.2.1 Hazard Identification. The final ISA hazard analysis tables are shown in SNF-4820. The main inventory of hazardous material in the ISA is the radionuclide content of the stored fuel. The toxicological hazards of the radionuclide inventory were reviewed. As described in Section D3.4.1.1, the radiological guidelines are more limiting than the toxicological guidelines for the release of SNF particulate. Other hazardous material identified by the hazard identification process includes pyrophoric metals and hydrides, sodium, oxidizers, hydrogen, diesel fuel, and other flammable or combustible materials. A specific and comprehensive analysis of all fire hazards associated with the ISA (SNF-4932) was completed to augment the standard hazard analysis.

The ISA does not have an operating history, so major hazards resulting from facility operation cannot be identified or summarized as suggested by DOE-STD-3009-94. However, as described in this section, the ISA spent fuel handling and storage activities are similar to those used by the independent spent fuel storage installations that were issued licenses under 10 CFR 72. These hazards include generation of combustible gases, failure of confinement barriers, defects in cask integrity, and the spread of external contamination.

D3.3.2.2 Hazard Classification. A final hazards categorization of the ISA facility was performed based on the final hazard analysis (SNF-4820) and accident analyses documentation for the facility. Consistent with DOE-STD-1027-92, the final categorization was based on the material-at-risk quantities identified in an individual cask inventory, as described in Section D3.4.1.1. The ISA material-at-risk quantities were compared against the DOE-STD-1027-92 threshold quantities. The ISA facility final hazard categorization found the ISA facility to be a Hazard Category 2 facility. This categorization level is consistent with the bases and guidance described in DOE-STD-1027-92.

The radioactive inventory for the 200 Area ISA is established in HNF-1755, *Initial Hazard Classification for the 200 Area Interim Storage Area, Project W-518*. HNF-1755 defines an inventory for safety analysis based on selecting the high-burnup fuel and fuel type that would result in the highest estimated dose to people exposed to the material, and then treating all the ISA fuel types as high-burnup fuel. The radioactive inventory for each of the ISA fuel types, as identified in HNF-1755, is presented in Tables D3-1, D3-2, and D3-3 for Fast Flux Test Facility
(FFTF) SNF; Neutron Radiography Facility (NRF) Training, Research and Isotope Production, General Atomics (TRIGA) SNF; and light water reactor (LWR) SNF, respectively.

The established safety basis radiological nuclide inventory of each fuel type was used to estimate the material quantities available for release by multiplying the material at risk and the specific nuclide inventories for each fuel type. The material at risk was taken as the amount of fuel stored in an individual cask for each fuel type. These quantities were compared against the Category 2 threshold values from DOE-STD-1027-92, Table A.1 (see Tables D3-1, D3-2 and D3-3). The sum of the material inventory/Category 2 quantity thresholds is 354 for the FFTF fuel, 0.03 for the NRF TRIGA fuel, and 82.2 for the commercial LWR fuel. Since the sum of ratios is greater than one when compared to Category 2 threshold criteria for both the FFTF and LWR fuel, the ISA remains a Category 2 facility as identified during the preliminary hazard classification.

**D3.3.2.3 Hazard Evaluation.** The final ISA facility hazard analysis identified hazards associated with handling and storage operations to be used in the ISA. Standard industrial hazards were identified and removed from the list of facility hazards used to identify the DBAs. The results of the hazard analysis identified hazard scenarios in the ISA. These scenarios were used to define the ISA DBAs selected for further analysis in Section D3.4.2.

The external hazards from human-generated threats to ISA operation identified in Section D1.6 involve only those from aircraft activity. Nine active airports within a 24-mile radius of the ISA (HNF-1786) define a maximum credible evaluation basis aircraft impact for the ISA. Therefore, the ISA facility is adequately protected from this external hazard.

Evaluation of the hazards of nearby external human-generated activities that could represent a threat to the facility is an NRC-equivalency requirement that is specified in HNF-SD-SNF-DB-003. Section D1.7 identifies and addresses the threats to the ISA from nearby external facilities. As stated in Section D1.6, other threats to ISA operation from external human-generated activities that are not currently well known at this time will be evaluated by the unreviewed safety question process.

The ISA fire hazard analysis (SNF-4932) identified the fire hazards, fire loading criteria, and appropriate requirements for addressing fire hazards during ISA operation based on applicable DOE Orders and regulations. The findings of the ISA fire hazard analysis were incorporated into the design and engineering of ISA SSCs and controls (e.g., fire loads, inspections, and watches). Therefore, credible fire hazards are addressed in the design of the facility. Threats from nearby facilities are identified in Section D1.7 based on applicable DOE Orders and regulations. Credible threats from nearby facilities are addressed in Section D1.7.
Table D3-1. Final Hazard Category Comparison for Fast Flux Test Facility Fuel Inventory per Cask at the 200 Area Interim Storage Area.

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>FFTF SNF inventory (single assembly)</th>
<th>FFTF SNF inventory (7 assemblies)</th>
<th>Category 2 threshold curies</th>
<th>Ratio of facility inventory to Category 2 curies</th>
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<tr>
<td>H-3</td>
<td>78.2</td>
<td>547</td>
<td>300,000</td>
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<td>Kr-85</td>
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<td>3,647</td>
<td>990,000</td>
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<td>Pu-238</td>
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</table>

Sum of the ratios: 353.55

**FFT**F = Fast Flux Test Facility.
**SNF** = Spent nuclear fuel.
Table D3-2. Final Hazard Category Comparison for NRF TRIGA Fuel Inventory per Cask at the 200 Area Interim Storage Area.

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>TRIGA SNF inventory (single assembly)</th>
<th>TRIGA SNF inventory (101 assemblies)</th>
<th>Category 3 threshold curies</th>
<th>Category 2 threshold curies</th>
<th>Ratio of facility inventory to Category 3 curies</th>
<th>Ratio of facility inventory to Category 2 curies</th>
</tr>
</thead>
<tbody>
<tr>
<td>H-3</td>
<td>0.005156</td>
<td>0.52</td>
<td>1.000</td>
<td>300,000</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>C-14</td>
<td>0.00001209</td>
<td>0.00</td>
<td>420</td>
<td>1,400,000</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>Fe-55</td>
<td>0.1904</td>
<td>19.23</td>
<td>5.400</td>
<td>11,000,000</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>Ni-59</td>
<td>0.0005488</td>
<td>0.06</td>
<td>11,800(1)</td>
<td>430,000</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>Ni-63</td>
<td>0.05943</td>
<td>6.00</td>
<td>5.400</td>
<td>4,500,000</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>Co-60</td>
<td>0.0006458</td>
<td>0.07</td>
<td>280</td>
<td>190,000</td>
<td>0.00</td>
<td>0.00</td>
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<tr>
<td>Zr-93</td>
<td>0.0002648</td>
<td>0.00</td>
<td>62</td>
<td>89,000</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>Kr-85</td>
<td>0.1273</td>
<td>12.86</td>
<td>20,000</td>
<td>28,000,000</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>Sr-90</td>
<td>1.864</td>
<td>188.26</td>
<td>16</td>
<td>22,000</td>
<td>11.77</td>
<td>0.009</td>
</tr>
<tr>
<td>Y-90</td>
<td>1.865</td>
<td>188.37</td>
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<td>430,000</td>
<td>0.13</td>
<td>0.000</td>
</tr>
<tr>
<td>Te-99</td>
<td>0.0004061</td>
<td>0.04</td>
<td>1,700</td>
<td>3,800,000</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>Ru-106</td>
<td>0.03391</td>
<td>3.42</td>
<td>100</td>
<td>6,500</td>
<td>0.03</td>
<td>0.001</td>
</tr>
<tr>
<td>Rh-106</td>
<td>0.03391</td>
<td>3.42</td>
<td>1,960(1)</td>
<td>430,000</td>
<td>0.00</td>
<td>0.000</td>
</tr>
<tr>
<td>Cd-113m</td>
<td>0.0001943</td>
<td>0.02</td>
<td>11.8(1)</td>
<td>430,000</td>
<td>0.00</td>
<td>0.000</td>
</tr>
<tr>
<td>Sb-125</td>
<td>0.01368</td>
<td>1.38</td>
<td>60(1)</td>
<td>430,000</td>
<td>0.02</td>
<td>0.000</td>
</tr>
<tr>
<td>Te-125m</td>
<td>0.0003337</td>
<td>0.34</td>
<td>36(1)</td>
<td>430,000</td>
<td>0.01</td>
<td>0.000</td>
</tr>
<tr>
<td>Cs-134</td>
<td>0.009104</td>
<td>0.92</td>
<td>42</td>
<td>60,000</td>
<td>0.02</td>
<td>0.000</td>
</tr>
<tr>
<td>Cs-137</td>
<td>1.965</td>
<td>198.47</td>
<td>60</td>
<td>89,000</td>
<td>3.31</td>
<td>0.002</td>
</tr>
<tr>
<td>Ce-144</td>
<td>0.1863</td>
<td>18.82</td>
<td>100</td>
<td>82,000</td>
<td>0.19</td>
<td>0.000</td>
</tr>
<tr>
<td>Pr-147</td>
<td>0.6730</td>
<td>67.97</td>
<td>1,000</td>
<td>840,000</td>
<td>0.07</td>
<td>0.000</td>
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<tr>
<td>Sn-151</td>
<td>0.04990</td>
<td>5.04</td>
<td>1,000</td>
<td>990,000</td>
<td>0.01</td>
<td>0.000</td>
</tr>
<tr>
<td>Eu-152</td>
<td>0.002756</td>
<td>0.28</td>
<td>200</td>
<td>130,000</td>
<td>0.00</td>
<td>0.000</td>
</tr>
<tr>
<td>Eu-154</td>
<td>0.002114</td>
<td>0.21</td>
<td>200</td>
<td>110,000</td>
<td>0.00</td>
<td>0.000</td>
</tr>
<tr>
<td>Eu-155</td>
<td>0.01505</td>
<td>1.52</td>
<td>940</td>
<td>730,000</td>
<td>0.00</td>
<td>0.000</td>
</tr>
<tr>
<td>U-234</td>
<td>0.002350</td>
<td>0.24</td>
<td>4.2</td>
<td>220</td>
<td>0.06</td>
<td>0.001</td>
</tr>
<tr>
<td>U-235</td>
<td>0.00007969</td>
<td>0.01</td>
<td>4.2</td>
<td>110,000,000</td>
<td>0.00</td>
<td>0.000</td>
</tr>
<tr>
<td>U-238</td>
<td>0.00005173</td>
<td>0.01</td>
<td>4.2</td>
<td>240</td>
<td>0.00</td>
<td>0.000</td>
</tr>
<tr>
<td>Np-237</td>
<td>0.000001769</td>
<td>0.00</td>
<td>0.42</td>
<td>58</td>
<td>0.00</td>
<td>0.000</td>
</tr>
<tr>
<td>Pu-238</td>
<td>0.0003155</td>
<td>0.03</td>
<td>0.62</td>
<td>62</td>
<td>0.05</td>
<td>0.000</td>
</tr>
<tr>
<td>Pu-239</td>
<td>0.004455</td>
<td>0.45</td>
<td>0.52</td>
<td>28</td>
<td>0.87</td>
<td>0.016</td>
</tr>
<tr>
<td>Pu-240</td>
<td>0.0002871</td>
<td>0.03</td>
<td>0.026(1)</td>
<td>55</td>
<td>1.15</td>
<td>0.001</td>
</tr>
<tr>
<td>Pu-241</td>
<td>0.0026373</td>
<td>0.27</td>
<td>32</td>
<td>2,900</td>
<td>0.01</td>
<td>0.000</td>
</tr>
<tr>
<td>Am-241</td>
<td>0.00005746</td>
<td>0.01</td>
<td>0.52</td>
<td>55</td>
<td>0.02</td>
<td>0.000</td>
</tr>
</tbody>
</table>

Sum totals | 20.72 | 0.030

(1) Category 3 threshold quantities for these radionuclides are not presented in DOE-STD-1027-92. These values are taken from WHC-SD-GN-HC-20002, 1996, Category 3 Threshold Quantities for Hazard Categorization of Nonreactor Facilities, Rev. 0.

SNF = spent nuclear fuel.

TRIGA = Training, Research and Isotope Production, General Atomics.
Table D3-3. Final Initial Hazard Category Comparison for Light Water Reactor Fuel Inventory per Cask at the 200 Area Interim Storage Area.

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>LWR fuel inventory (single assembly)</th>
<th>Category 2 threshold curies</th>
<th>Ratio of facility inventory to Category 2 curies</th>
</tr>
</thead>
<tbody>
<tr>
<td>H-3</td>
<td>92.8</td>
<td>300,000</td>
<td>0.00</td>
</tr>
<tr>
<td>Co-60</td>
<td>788</td>
<td>190,000</td>
<td>0.00</td>
</tr>
<tr>
<td>Kr-85</td>
<td>1,310</td>
<td>28,000,000</td>
<td>0.00</td>
</tr>
<tr>
<td>Sr-90</td>
<td>19,800</td>
<td>22,000</td>
<td>0.90</td>
</tr>
<tr>
<td>Y-90</td>
<td>19,800</td>
<td>430,000</td>
<td>0.05</td>
</tr>
<tr>
<td>Tc-99</td>
<td>5.21</td>
<td>3,800,000</td>
<td>0.00</td>
</tr>
<tr>
<td>Rh-106</td>
<td>8.43</td>
<td>430,000</td>
<td>0.00</td>
</tr>
<tr>
<td>Ru-106</td>
<td>8.42</td>
<td>6,500</td>
<td>0.00</td>
</tr>
<tr>
<td>Cd-113m</td>
<td>13.1</td>
<td>430,000</td>
<td>0.00</td>
</tr>
<tr>
<td>Sb-125</td>
<td>130</td>
<td>430,000</td>
<td>0.00</td>
</tr>
<tr>
<td>Tc-125m</td>
<td>31.7</td>
<td>430,000</td>
<td>0.00</td>
</tr>
<tr>
<td>I-129</td>
<td>0.0126</td>
<td>430,000</td>
<td>0.00</td>
</tr>
<tr>
<td>Cs-134</td>
<td>491</td>
<td>60,000</td>
<td>0.81</td>
</tr>
<tr>
<td>Cs-137</td>
<td>28,500</td>
<td>89,000</td>
<td>0.32</td>
</tr>
<tr>
<td>Pr-147</td>
<td>817</td>
<td>840,000</td>
<td>0.00</td>
</tr>
<tr>
<td>Sm-151</td>
<td>123</td>
<td>990,000</td>
<td>0.00</td>
</tr>
<tr>
<td>Eu-154</td>
<td>1.550</td>
<td>110,000</td>
<td>0.01</td>
</tr>
<tr>
<td>Eu-155</td>
<td>402</td>
<td>730,000</td>
<td>0.00</td>
</tr>
<tr>
<td>U-234</td>
<td>0.472</td>
<td>220</td>
<td>0.00</td>
</tr>
<tr>
<td>U-235</td>
<td>0.00691</td>
<td>240</td>
<td>0.00</td>
</tr>
<tr>
<td>U-236</td>
<td>0.107</td>
<td>55</td>
<td>0.00</td>
</tr>
<tr>
<td>U-238</td>
<td>0.130</td>
<td>240</td>
<td>0.00</td>
</tr>
<tr>
<td>Np-237</td>
<td>0.134</td>
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<td>0.00</td>
</tr>
<tr>
<td>Np-239</td>
<td>12.8</td>
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<td>0.23</td>
</tr>
<tr>
<td>Pu-238</td>
<td>1.360</td>
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<td>21.94</td>
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<td>4.46</td>
</tr>
<tr>
<td>Pu-240</td>
<td>212</td>
<td>55</td>
<td>3.85</td>
</tr>
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<td>Pu-241</td>
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<td>2,900</td>
<td>8.41</td>
</tr>
<tr>
<td>Pu-242</td>
<td>0.939</td>
<td>55</td>
<td>0.02</td>
</tr>
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<td>Am-241</td>
<td>944</td>
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<td>17.16</td>
</tr>
<tr>
<td>Am-242</td>
<td>10.7</td>
<td>55</td>
<td>0.19</td>
</tr>
<tr>
<td>Am-242m</td>
<td>10.7</td>
<td>56</td>
<td>0.19</td>
</tr>
<tr>
<td>Am-243</td>
<td>12.8</td>
<td>55</td>
<td>0.23</td>
</tr>
<tr>
<td>Cm-242</td>
<td>8.82</td>
<td>1,700</td>
<td>0.01</td>
</tr>
<tr>
<td>Cm-243</td>
<td>10.5</td>
<td>55</td>
<td>0.19</td>
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<tr>
<td>Cm-244</td>
<td>1,210</td>
<td>55</td>
<td>22.00</td>
</tr>
</tbody>
</table>

Sum of the ratios: 82.17

LWR = light water reactor.
D3.3.2.3.1 Planned Design and Operational Safety Improvements. This section discusses commitments for planned, but not yet implemented, major design, handling, or storage improvements for the facility. However, there are currently no outstanding major improvements for improved safety identified or planned as a result of the hazard evaluation that are not part of the current design and planned facility operations.

D3.3.2.3.2 Defense in Depth. A summary of fundamental points relevant to the concept of defense in depth are described in Section 3.3.2.3.2 of the SNF Project FSAR.

Features Chosen to Provide Defense in Depth for the 200 Area Interim Storage Area

Defense-in-depth features for the ISA were selected based on a relative ranking of the hazards from the hazard identification process, followed by selection of the safety-class and safety-significant features and TSRs for the DBAs, which are described in Section D3.4.2. Preventive and mitigative features identified in the hazard analysis (SNF-4820) but not identified in the accident analysis as safety-class, safety-significant, or TSRs are identified as additional defense-in-depth features. The defense-in-depth features are presented in the tables that accompany each DBA in Section D3.4.2. The NRF TRIGA cask and DOT-6M container would be upgraded to Safety Significant for their significant contribution to defense-in-depth, but have already been designated Safety Significant based on the NRC important-to-safety classification. Administrative features identified in these tables are in addition to those already identified in Chapters D4.0 and D5.0, and the programmatic chapters of this Annex (e.g., Chapters D7.0, D8.0, and D11.0).

All SSCs are designed in accordance with applicable codes and standards with a high degree of reliability and simplicity, and the design encompasses human factors considerations to ensure that operations can be conducted safely. Defense-in-depth features for preventing and mitigating hazards and accidents have also been identified. An abnormal or accident cask is defined as one received out of specification or one that is dropped, impacted, or damaged at the ISA. Such casks will be handled using recovery operations under operations-related procedures. Recovery will be based on analysis and the development of a recovery plan by an appropriately qualified recovery team.

Safety-Significant Structures, Systems, and Components

Safety-significant SSCs are predominantly required to prevent or mitigate consequences of postulated accident events to the collocated on-site worker. In addition, DOE-STD-3009-94 suggests that SSCs be designated as Safety Significant, if they play a key role in defense in depth or worker safety. The severity of the event being prevented or mitigated, and the number of barriers present, are provided in DOE-STD-3009-94 as guidance for the identification of defense-in-depth safety-significant SSCs.
Technical Safety Requirements

TSRs were identified for postulated accident events that could challenge accident consequence release limits and evaluation guidelines for the off-site public and collocated on-site worker. These TSRs are identified in the individual DBA sections and further explained in Chapter D5.0. In addition, criticality prevention features are controlled by the TSRs, as identified in Chapter 5.0 of the SNF Project FSAR and in Chapters D5.0 and D6.0. No defense-in-depth features are identified as requiring TSR coverage.

D3.3.2.3.3 Worker Safety. Worker safety for the ISA is ensured by a combination of design features that reduce exposure to radioactive, toxic, and industrial hazards, and by institutional practices that, in total, provide protection of workers from these hazards. Protection of the facility worker from the standard industrial hazards identified for the ISA is achieved through adherence to the institutional safety programs described in Chapters 7.0, 8.0, 9.0, 11.0, 15.0, and 17.0 of the SNF Project FSAR and documented in lower-tier documents (e.g., health and safety plans and job hazards analyses). These industrial hazards do not require specific safety-significant SSCs or TSR-level administrative features. Therefore, in accordance with the guidance of DOE-STD-3009-94, the remainder of this section deals with protecting workers from the hazards of facility operation that are exclusive of standard industrial hazards.

The final ISA hazard analysis provides an overview of the major features that protect facility workers at the ISA (SNF-4820). Worker safety features are an integral part of facility design and operation. The major features of worker protection are identified in Table D3-4 and are categorized by hazard. The features presented in Table D3-4 are in addition to those identified as safety-class or safety-significant features in the DBA sections. The hazard energy source or material and hazardous condition are identified along with protective worker safety features, which include passive, active, and administrative features. SSCs are identified as Safety Significant for the ISA based on their ability to prevent or mitigate serious impacts to worker safety. The NRF TRIGA cask and DOT-6M container would be upgraded to Safety Significant for their significant contribution to worker safety, but have already been designated Safety Significant based on the NRC important-to-safety classification.

D3.3.2.3.4 Environmental Protection. The hazard to the environment from ISA operations involves the potential release of contaminants. The release pathway for these contaminants is only via the air to the boundaries and receptors discussed in Section D1.3.1.3. No liquid release hazards or accidents have been identified, and no contaminant releases to the ground or groundwater are involved for the ISA.

Based on the ISA design and operating information, no use of toxic chemicals has been identified. The toxicological hazards of the radionuclide inventory have been reviewed. As described in Section D3.4.1.1, the radiological guidelines are more limiting than the toxicological guidelines for any release from the material stored in the ISA. Implementation of the prevention and mitigation features will prevent large releases that could have significant environmental impact.
<table>
<thead>
<tr>
<th>Location/checklist entry</th>
<th>Hazardous condition</th>
<th>Potential accident</th>
<th>Worker consequences</th>
<th>Features in addition to those identified in the design basis accident sections</th>
</tr>
</thead>
<tbody>
<tr>
<td>Storage area/ B-07 B-10 B-11</td>
<td>Cask overheating Hot surfaces Hot gases</td>
<td>Exceeding the design temperature of the cask Loss of confinement of the cask Pressurization of the cask</td>
<td>Worker exposure to radioactive particulate Personnel injury</td>
<td>Passively cooled cask design; cask designed to withstand temperatures in excess of any of those anticipated. Designed for internal pressure to withstand worst possible scenario. Cooling channel in ISC.</td>
</tr>
<tr>
<td>Storage area/ D-02 D-06</td>
<td>Loss of shipping container structural integrity</td>
<td>Drop of container upon lifting</td>
<td>Personnel injury</td>
<td>Paint coating on container. NAC-1 channels on bottom of container. Routine inspection and maintenance.</td>
</tr>
<tr>
<td>Storage area/ F-01 F-02 F-04 F-05</td>
<td>Vehicle collision with cask Vehicle collision with other equipment</td>
<td>Vehicle collision with:  - Casks  - Equipment  - Obstructions. Loss of confinement</td>
<td>Worker exposure to radioactive particulate</td>
<td>Cask designed to withstand impact. Fenced perimeter and locked gates. Restricted access inside fenced area.</td>
</tr>
<tr>
<td>Location/checklist entry</td>
<td>Hazardous condition</td>
<td>Potential accident</td>
<td>Worker consequences</td>
<td>Engineering features</td>
</tr>
<tr>
<td>--------------------------</td>
<td>--------------------</td>
<td>--------------------</td>
<td>---------------------</td>
<td>---------------------</td>
</tr>
<tr>
<td>Storage area/ G-03 G-06 G-13</td>
<td>A drop of the boom on the cask</td>
<td>Cask is breached</td>
<td>Personnel injury</td>
<td>Hoist designed for single failure.</td>
</tr>
<tr>
<td></td>
<td>A drop of the cask while being moved by the crane</td>
<td>Cask falls from crane to pad or transporter</td>
<td>Worker exposure to radioactive particulate</td>
<td>ISC survives 4-ft drop to unyielding surface.</td>
</tr>
<tr>
<td></td>
<td>Cask is overturned</td>
<td>Cask is overturned</td>
<td></td>
<td>DOT-6M cask.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>TRIGA cask designed to withstand a 109-in. drop to concrete.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>For NAC-1 cask, upper and lower impact limiters.</td>
</tr>
<tr>
<td>Storage area/ H-06 H-07 H-11</td>
<td>Overpressurization of cask and release of gases from cask</td>
<td>Loss of confinement and release of radioactive gases and particulate</td>
<td>Personnel injury</td>
<td>Casks designed to withstand pressurization.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Worker exposure to radioactive particulate</td>
<td>Annular seal pressure is greater than cavity pressure or external pressure, so any leak would not release radioactive gas.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Personnel contamination</td>
<td></td>
</tr>
<tr>
<td>Storage area/ J-06 L-11</td>
<td>Flammable hydrogen-air mixture and an ignition source causing combustion</td>
<td>Deflagration may cause excessive pressure or cask damage</td>
<td>Personnel injury</td>
<td>Casks are sealed to prevent ingress of water.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Worker exposure to radioactive particulate</td>
<td>Fuel is dried before storing.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Personnel contamination</td>
<td></td>
</tr>
<tr>
<td>Storage area/ J-11 M-01</td>
<td>Pyrophoric material exposed to air and/or water with sufficient surface area and temperature</td>
<td>Pyrophoric sodium metal reacts upon cladding breach</td>
<td>Personnel injury</td>
<td>Structural design of cask (double seals) is such that breaches have a very low probability of occurring.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Contamination release</td>
<td>Air ingress into cask would be limited by a small breach.</td>
</tr>
</tbody>
</table>
Table D3-4. Safety Features for Facility Workers (S1 Consequence Items). (4 sheets)

<table>
<thead>
<tr>
<th>Location/checklist entry</th>
<th>Hazardous condition</th>
<th>Potential accident</th>
<th>Worker consequences</th>
<th>Features in addition to those identified in the design basis accident sections</th>
</tr>
</thead>
<tbody>
<tr>
<td>Storage area/ L-01</td>
<td>Fire inside the ISA fence</td>
<td>Worker exposed to high heat from fire</td>
<td>Personnel injury</td>
<td>Support buildings and pad materials selected to minimize flammability. Casks can withstand design basis fires. Intact cladding.</td>
</tr>
<tr>
<td>L-02</td>
<td></td>
<td></td>
<td></td>
<td>Flammable material inventories controlled within the fenced area.</td>
</tr>
<tr>
<td>L-03</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>L-07</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>L-14</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>L-16</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Storage area/ P-01</td>
<td>External explosion resulting in a shock wave introduces dust, heat, and/or projectiles into the storage area. Radioactive material, toxic material, and direct radiation are present.</td>
<td>Shock wave impacts personnel. Personnel are hit by debris Personnel are exposed to heat from explosion</td>
<td>Personnel injury Worker exposure to radioactive particulate</td>
<td>ISA is isolated from major roads, reducing the possibility of explosions outside the facility affecting the ISA. Distance of the ISA from other facilities reduces the consequences of external events.</td>
</tr>
<tr>
<td>P-03</td>
<td></td>
<td></td>
<td></td>
<td>Facility is addressed by the fire hazards analysis.</td>
</tr>
</tbody>
</table>
Table D3-4. Safety Features for Facility Workers (S1 Consequence Items). (4 sheets)

<table>
<thead>
<tr>
<th>Location/checklist entry</th>
<th>Hazardous condition</th>
<th>Potential accident</th>
<th>Worker consequences</th>
<th>Features in addition to those identified in the design basis accident sections</th>
</tr>
</thead>
<tbody>
<tr>
<td>Storage area/</td>
<td>Forces exerted on the facility from lateral and horizontal accelerations</td>
<td>Damage to or failure of structures, systems, or components ISA flooded Natural energy sources</td>
<td>Personnel injury</td>
<td>ISA facility structure and confinement components are built to appropriate seismic criteria, as documented in the design. ISA facility is designed for the lightning hazards. The facility structure is designed to withstand natural phenomena hazard loads.</td>
</tr>
<tr>
<td>R-01</td>
<td></td>
<td></td>
<td></td>
<td>Personnel are trained in sitewide and facility-specific emergency responses.</td>
</tr>
<tr>
<td>R-02</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>R-03</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>R-04</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>R-05</td>
<td>ISA flooded</td>
<td>ISA flooded</td>
<td>Personnel injury</td>
<td></td>
</tr>
<tr>
<td>R-06</td>
<td>Natural energy sources</td>
<td>Natural energy sources</td>
<td>Personnel injury</td>
<td></td>
</tr>
<tr>
<td>R-07</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>R-08</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>R-09</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>R-10</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

DOT = U.S. Department of Transportation.
FFT = Fast Flux Test Facility.
ISA = interim storage area.
ISC = interim storage cask.

LWR = light water reactor.
NAC = Nuclear Assurance Corporation.
TRIGA = Training, Research and Isotope Production, General Atomics.
TSR = Technical Safety Requirement.
The project features that protect the on-site collocated worker and the off-site public against radiological exposure also serve to prevent and mitigate radiological release to the environment. In addition, sitewide programs for environmental monitoring provide for assessment of the impact of facility releases. Normal ISA handling or storage activities are expected to have a minor impact on the local and regional environment.

**D3.3.2.3.5 Accident Selection.** The methodology for the selection of DBAs is specified in DOE-STD-3009-94. DBAs are to be selected so that the range of accident scenarios analyzed in the accident analysis represents a complete set of representative and bounding conditions. This is a common requirement among the SNF Project facilities and is described in Section 3.3.2.3.5 of the SNF Project FSAR. The list of candidate accidents resulting from the hazards binning process for the ISA facility is presented in Table D3-5. The table contains all identified Category S3 and S2 events. All of these events, and the controls selected for their prevention and mitigation, are described in Sections D3.4.2.1 through D3.4.2.7.

Table D3-5. Binned Listing of Candidates Sorted By Risk Ranking for 200 Area Interim Storage Area.

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Release energy</th>
<th>Risk ranking</th>
<th>Reference designator</th>
</tr>
</thead>
<tbody>
<tr>
<td>D3.4.2.1 Handling/drop</td>
<td>low</td>
<td>5</td>
<td>G-03 G-06 G-13</td>
</tr>
<tr>
<td>D3.4.2.2 Mobile crane fall</td>
<td>low</td>
<td>5</td>
<td>G-03</td>
</tr>
<tr>
<td>D3.4.2.3 Cask tipover</td>
<td>low</td>
<td>5</td>
<td>G-03 G-06 G-13</td>
</tr>
<tr>
<td>D3.4.2.4 Fuel rod rupture</td>
<td>low</td>
<td>5</td>
<td>H-06 H-11</td>
</tr>
<tr>
<td>D3.4.2.5 Seismic</td>
<td>med/low</td>
<td>7</td>
<td>R-01</td>
</tr>
<tr>
<td>D3.4.2.6 Tornado/wind</td>
<td>med/low</td>
<td>7</td>
<td>R-06 R-08</td>
</tr>
<tr>
<td>D3.4.2.7 Fire</td>
<td>medium</td>
<td>5</td>
<td>L-03 L-07</td>
</tr>
<tr>
<td>D6.0 Inadvertent nuclear</td>
<td>high</td>
<td>6</td>
<td>K-01 K-02 K-05 K-10 K-12 K-15</td>
</tr>
<tr>
<td>criticality</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
D3.3.3 Abnormal Events for the 200 Area Interim Storage Area Facility

Abnormal events are operating conditions resulting from situations outside of normal operations, where normal operations are defined by operation and maintenance procedures. These abnormal events encompass malfunctions of systems, operating upset conditions, or operator error. Abnormal events can be expected to occur annually or several times during the lifetime of the facility.

Abnormal events may impact operational or programmatic schedules; however, the consequences of the events are near zero or are standard industrial hazards that may include worker radiation exposure. Events having radiological consequences greater than allowed by the facility radiological protection and ALARA (as low as reasonably achievable) programs do not fit the abnormal event profile and are required to be analyzed as accidents by the DOE safety analysis process.

D3.4 ACCIDENT ANALYSIS

This section presents the methodology used to develop the DBAs identified in Section D3.3 and the results of the quantification of the consequences of those events. Also presented are the safety-class and safety-significant SSCs and TSRs related to these accident events that are necessary for protection of the off-site public and on-site workers. For each DBA, the following standard topics are discussed:

- Scenario development
- Source term analysis
- Consequence analysis
- Comparison to guidelines
- Summary of safety SSCs and TSRs.

D3.4.1 Methodology

This section identifies any ISA-specific methods, assumptions, or methodology used to quantify the consequences of the DBAs. The methods, assumptions, or methodology used to quantify the consequences of DBAs that are common or generic to all SNF Project facilities at the K Basins, Cold Vacuum Drying Facility, Canister Storage Building, and 200 Area ISA are described in Chapter 3.0 of the SNF Project FSAR.

D3.4.1.1 Source Term. Hypothetical release source terms are developed for each fuel type in this section. The 200 Area ISA inventory assumed for the analysis is based on a per cask basis and either includes seven FFTF interim storage cask (ISC) assemblies, all 101 TRIGA elements, or one LWR assembly.
For this report, values for the "inhalation dose factor" were taken from Federal Guidance Report Number 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation* (EPA 1988). The computed committed dose for each radionuclide identified in the individual fuel types is calculated in Tables D3-6, D3-7, and D3-8. To indicate the relative importance of the various radionuclides to the total dose, the last column shows the fraction contributed by each nuclide to the total dose per gram inhaled. As can be seen from the table, the radionuclides that account for a majority of the unit doses are $^{238}$Pu, $^{239}$Pu, $^{240}$Pu, $^{241}$Pu, $^{242}$Am, and $^{244}$Cm.

Note that the unit dose value presented in these tables assumes the uniform mixture of spent fuel is present in the airborne material. In other words, all nuclides have the same release fraction. The hypothetical accident scenarios have a preferential release of gaseous nuclides and require summing of the contributions of individual radionuclides.

The SNF is primarily uranium oxide, and mixed uranium and plutonium oxide, which is known to have toxicological effects. However, the toxicological consequences of the release of these substances will not require additional mitigating features beyond those already required by the radiological doses.

**D3.4.1.2 Consequence Analysis.** For each SNF type, there is a bounding credible accident for which a maximum credible radiological release and dose consequence is determined. These radiological doses to a maximum duration exposed on-site and off-site receptor are estimated by using the following equation:

$$D = M \times \left[ \frac{X}{Q'} \times BR \times UD \right]$$

where

- $D$ = committed effective dose equivalent (CEDE) (Sv [rem])
- $M$ = unit mass of respirable material released (radionuclide mass per assembly)
- $X/Q'$ = time-integrated atmospheric transport factor (s/m³)
- $BR$ = breathing rate (m³/s)
- $UD$ = dose per unit respirable radioactive material inhaled (Sv/g [rem/assembly]).

The quantity of respirable material released ($M$) is determined by the specific accident scenario (e.g., crane fall or cask drop) for each accident. The parameters and values of $X/Q'$ and breathing rate ($BR$) are defined and specified in their respective analyses. The accident-specific, location-specific, release fraction and fuel type-specific on-site and off-site dose consequences are estimated from this formulation.
Table D3-6. Fast Flux Test Facility Radionuclide Composition of Fuel and the Unit Dose Factor per Assembly. (2 sheets)

<table>
<thead>
<tr>
<th>Fission and Activation Products</th>
<th>Activity (Ci/Asmbl)</th>
<th>Activity (Bq/Asmbl)</th>
<th>Inhalation dose factor (Sv/Bq)</th>
<th>Class</th>
<th>Unit dose (Sv/Asmbl)</th>
<th>Percent of Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>H-3</td>
<td>7.82E+01</td>
<td>2.89E+12</td>
<td>2.60E-11</td>
<td>Vapor</td>
<td>7.51E+01</td>
<td>0.00%</td>
</tr>
<tr>
<td>C-14</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>5.64E-10</td>
<td>Organic</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Fe-55</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>7.26E-10</td>
<td>D</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Co-60</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>8.94E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Ni-59</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>7.31E-10</td>
<td>Vapor</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Ni-63</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>1.70E-09</td>
<td>Vapor</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Se-79</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>2.66E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Kr-85</td>
<td>6.74E+02</td>
<td>2.49E+13</td>
<td>3.57E-13</td>
<td>Gas(40)</td>
<td>8.90E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sr-90</td>
<td>5.04E+03</td>
<td>1.86E+14</td>
<td>6.47E-08</td>
<td>D</td>
<td>1.21E+07</td>
<td>0.19%</td>
</tr>
<tr>
<td>Y-90</td>
<td>5.04E+03</td>
<td>1.86E+14</td>
<td>2.13E-09</td>
<td>W</td>
<td>3.97E+05</td>
<td>0.01%</td>
</tr>
<tr>
<td>Zr-93</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>8.67E-08</td>
<td>D</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Nb-93m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>8.68E-10</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Tc-99</td>
<td>1.99E+00</td>
<td>7.36E+10</td>
<td>2.25E-09</td>
<td>W</td>
<td>1.66E+02</td>
<td>0.00%</td>
</tr>
<tr>
<td>Ru-106</td>
<td>4.89E+03</td>
<td>1.81E+14</td>
<td>3.18E-08</td>
<td>W</td>
<td>5.75E+06</td>
<td>0.09%</td>
</tr>
<tr>
<td>Rh-106</td>
<td>4.89E+03</td>
<td>1.81E+14</td>
<td>DP</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Pd-107</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>2.19E-10</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Ag-110</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>DP</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Ag-110m</td>
<td>8.89E+00</td>
<td>3.29E+11</td>
<td>8.34E-09</td>
<td>W</td>
<td>2.74E+03</td>
<td>0.00%</td>
</tr>
<tr>
<td>Cd-113m</td>
<td>1.29E+01</td>
<td>4.77E+11</td>
<td>4.13E-07</td>
<td>D</td>
<td>1.97E+05</td>
<td>0.00%</td>
</tr>
<tr>
<td>In-113m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>1.11E-11</td>
<td>D</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sn-113</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>2.88E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sn-119m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>1.69E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sn-121m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>3.11E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sn-123</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>8.79E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sn-126</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>2.69E-08</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sb-125</td>
<td>1.21E+03</td>
<td>4.48E+13</td>
<td>3.30E-09</td>
<td>W</td>
<td>1.48E+05</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sb-126</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>3.17E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sb-126m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>9.17E-12</td>
<td>D</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Te-123m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>2.86E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Te-125m</td>
<td>2.96E+02</td>
<td>1.10E+13</td>
<td>1.97E-09</td>
<td>W</td>
<td>2.16E+04</td>
<td>0.00%</td>
</tr>
<tr>
<td>Te-127</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>8.60E-11</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Te-127m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>5.81E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>I-129</td>
<td>6.63E-03</td>
<td>2.45E+08</td>
<td>4.69E-08</td>
<td>D</td>
<td>1.15E+01</td>
<td>0.00%</td>
</tr>
<tr>
<td>Cs-134</td>
<td>2.51E+03</td>
<td>9.29E+13</td>
<td>1.25E-08</td>
<td>D</td>
<td>1.16E+06</td>
<td>0.02%</td>
</tr>
<tr>
<td>Cs-135</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>1.23E-09</td>
<td>D</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Cs-137</td>
<td>1.37E+04</td>
<td>5.07E+14</td>
<td>8.63E-09</td>
<td>D</td>
<td>4.37E+06</td>
<td>0.07%</td>
</tr>
<tr>
<td>Ba-137m</td>
<td>1.30E+04</td>
<td>4.80E+14</td>
<td>DP</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
</tbody>
</table>
### Table D3-6. Fast Flux Test Facility Radionuclide Composition of Fuel and the Unit Dose Factor per Assembly. (2 sheets)

<table>
<thead>
<tr>
<th>Fission and Activation Products</th>
<th>Activity (Ci/Asmbl)</th>
<th>Activity (Bq/Asmbl)</th>
<th>Inhalation dose factor (Sv/Bq)</th>
<th>Class</th>
<th>Unit dose (Sv/Asmbl)</th>
<th>Percent of Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ce-144</td>
<td>1.66E+03</td>
<td>6.14E+13</td>
<td>5.84E-08</td>
<td>W</td>
<td>3.59E+06</td>
<td>0.06%</td>
</tr>
<tr>
<td>Pr-144</td>
<td>1.66E+03</td>
<td>6.14E+13</td>
<td>1.10E-11</td>
<td>W</td>
<td>6.76E+02</td>
<td>0.00%</td>
</tr>
<tr>
<td>Pr-144m</td>
<td>2.37E+01</td>
<td>8.78E+11</td>
<td>DP</td>
<td></td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Pm-147</td>
<td>8.73E+03</td>
<td>3.23E+14</td>
<td>6.97E-09</td>
<td>W</td>
<td>2.25E+06</td>
<td>0.04%</td>
</tr>
<tr>
<td>Sm-151</td>
<td>5.21E+02</td>
<td>1.93E+13</td>
<td>8.10E-09</td>
<td>W</td>
<td>1.56E+05</td>
<td>0.00%</td>
</tr>
<tr>
<td>Eu-152</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>5.97E-08</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Eu-154</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>7.73E-08</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Eu-155</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>1.12E-08</td>
<td>W</td>
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<td>0.00%</td>
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<tr>
<td>Gd-153</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
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<td>D</td>
<td>0.00E+00</td>
<td>0.00%</td>
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<tr>
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<td>2.36E+15</td>
<td>9.83E-07</td>
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<td>3.01E+07</td>
<td>0.48%</td>
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<td>U-234</td>
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<td>1.97E-06</td>
<td>W</td>
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<td>0.00%</td>
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<tr>
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<td>2.01E-06</td>
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<tr>
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<td>1.91E-06</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Np-237</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>1.46E-04</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
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<td>2.81E+02</td>
<td>1.04E+13</td>
<td>1.06E-04</td>
<td>W</td>
<td>1.10E+09</td>
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<td>1.29E+13</td>
<td>1.16E-04</td>
<td>W</td>
<td>1.50E+09</td>
<td>24.00%</td>
</tr>
<tr>
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<td>9.92E+12</td>
<td>1.16E-04</td>
<td>W</td>
<td>1.15E+09</td>
<td>18.43%</td>
</tr>
<tr>
<td>Pu-241</td>
<td>1.46E+04</td>
<td>5.40E+14</td>
<td>2.23E-06</td>
<td>W</td>
<td>1.20E+09</td>
<td>19.30%</td>
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<td>0.00E+00</td>
<td>1.11E-04</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
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<td>9.36E+12</td>
<td>1.20E-04</td>
<td>W</td>
<td>1.12E+09</td>
<td>18.00%</td>
</tr>
<tr>
<td>Am-242</td>
<td>1.92E+01</td>
<td>7.11E+11</td>
<td>1.58E-08</td>
<td>W</td>
<td>1.12E+04</td>
<td>0.00%</td>
</tr>
<tr>
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<td>1.93E+01</td>
<td>7.14E+11</td>
<td>1.15E-04</td>
<td>W</td>
<td>8.21E+07</td>
<td>1.32%</td>
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<td>1.19E-04</td>
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<td>0.00%</td>
</tr>
<tr>
<td>Cm-242</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>4.67E-06</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Cm-244</td>
<td>2.04E+01</td>
<td>7.55E+11</td>
<td>6.70E-05</td>
<td>W</td>
<td>5.06E+07</td>
<td>0.81%</td>
</tr>
<tr>
<td>Subtotal:</td>
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<td>5.85E+14</td>
<td>1.03E-03</td>
<td></td>
<td>6.21E+09</td>
<td>99.52%</td>
</tr>
</tbody>
</table>

| Total:                        | 6.24E+09            |                     |                               |       |                     |                 |

---


(2) Internal dose factor for tritium was increased by 50% to include skin absorption.

(3) The unit dose is the product of the normalized activity and the inhalation dose factor. To convert Sv/g to rem/g, multiply by 100.

(4) Kr-85 is a noble gas. It does not accumulate in the body; therefore, its internal dose factor is zero. The value shown is the external dose rate factor for submersion in an infinite cloud divided by the light activity breathing rate.

DP = daughter products (included with parents and not tracked individually).
Table D3-7. NRF TRIGA Radionuclide Composition of Fuel and the Unit Dose Factor per Total Inventory (101 elements). (2 sheets)

<table>
<thead>
<tr>
<th>Fission and Activation Products</th>
<th>Activity (Ci/Asmbl)</th>
<th>Activity (Bq/Asmbl)</th>
<th>Inhalation dose factor (Sv/Bq)</th>
<th>Class</th>
<th>Unit dose (Sv/Asmbl)</th>
<th>Percent of Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>H-3</td>
<td>5.16E+03</td>
<td>1.91E+08</td>
<td>2.60E-11</td>
<td>Vapor</td>
<td>4.95E+03</td>
<td>0.00%</td>
</tr>
<tr>
<td>C-14</td>
<td>1.21E+05</td>
<td>4.47E+05</td>
<td>5.64E-10</td>
<td>Organic</td>
<td>2.52E-04</td>
<td>0.00%</td>
</tr>
<tr>
<td>Fe-55</td>
<td>1.90E-01</td>
<td>7.04E+09</td>
<td>7.26E-10</td>
<td>D</td>
<td>5.11E+00</td>
<td>0.02%</td>
</tr>
<tr>
<td>Co-60</td>
<td>6.46E-04</td>
<td>2.39E+07</td>
<td>8.94E-09</td>
<td>W</td>
<td>2.14E+01</td>
<td>0.00%</td>
</tr>
<tr>
<td>Ni-59</td>
<td>5.49E-04</td>
<td>2.03E+07</td>
<td>7.31E-10</td>
<td>Vapor</td>
<td>1.48E-02</td>
<td>0.00%</td>
</tr>
<tr>
<td>Ni-63</td>
<td>5.94E-02</td>
<td>2.20E+09</td>
<td>1.70E-09</td>
<td>Vapor</td>
<td>3.74E+00</td>
<td>0.01%</td>
</tr>
<tr>
<td>Se-79</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>2.66E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Kr-85</td>
<td>1.27E-01</td>
<td>4.71E+09</td>
<td>3.57E-13</td>
<td>Gas⁴⁰</td>
<td>1.68E-03</td>
<td>0.00%</td>
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<tr>
<td>Sr-90</td>
<td>1.86E+00</td>
<td>6.90E+10</td>
<td>6.47E-08</td>
<td>D</td>
<td>4.46E+03</td>
<td>15.84%</td>
</tr>
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<td>6.90E+10</td>
<td>2.13E-09</td>
<td>W</td>
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<td>0.52%</td>
</tr>
<tr>
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<td>8.67E-08</td>
<td>D</td>
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<td>0.00%</td>
</tr>
<tr>
<td>Nb-93m</td>
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<td>8.68E-10</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
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<tr>
<td>Tc-99</td>
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<td>2.25E-09</td>
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<td>3.38E-02</td>
<td>0.00%</td>
</tr>
<tr>
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<tr>
<td>Rh-106</td>
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<td>DP</td>
<td></td>
<td>0.00E+00</td>
<td>0.00%</td>
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<td>Pd-107</td>
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<td>0.00E+00</td>
<td>2.19E-10</td>
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<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Ag-110m</td>
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<td>0.00E+00</td>
<td>DP</td>
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<td>0.00E+00</td>
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</tr>
<tr>
<td>Ag-110m</td>
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<tr>
<td>Cd-113m</td>
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<td>7.19E+06</td>
<td>4.13E-07</td>
<td>D</td>
<td>2.97E+00</td>
<td>0.01%</td>
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<tr>
<td>In-113m</td>
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<td>0.00E+00</td>
<td>1.11E-11</td>
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<tr>
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<td>0.00E+00</td>
<td>2.88E-09</td>
<td>W</td>
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<td>0.00%</td>
</tr>
<tr>
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<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sn-121m</td>
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<td>0.00E+00</td>
<td>3.11E-09</td>
<td>W</td>
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<td>0.00%</td>
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<tr>
<td>Sn-123</td>
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<td>8.79E-09</td>
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<td>0.00%</td>
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<tr>
<td>Sn-126</td>
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<td>2.69E-08</td>
<td>W</td>
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<tr>
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</tr>
<tr>
<td>Sb-126</td>
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<tr>
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<tr>
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<tr>
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</tr>
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<td>1.10E-11</td>
<td>W</td>
<td>7.58E-02</td>
<td>0.00%</td>
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</table>
Table D3-7. NRF TRIGA Radionuclide Composition of Fuel and the Unit Dose Factor per Total Inventory (101 elements). (2 sheets)

<table>
<thead>
<tr>
<th>Fission and Activation Products</th>
<th>Activity (Ci/Asmbl)</th>
<th>Activity (Bq/Asmbl)</th>
<th>Inhalation dose factor (Sv/Bq)</th>
<th>Class</th>
<th>Unit dose (Sv/Asmbl)</th>
<th>Percent of Total</th>
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<td>6.97E-09</td>
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<td>1.74E+02</td>
<td>0.62%</td>
</tr>
<tr>
<td>Sm-151</td>
<td>4.99E-02</td>
<td>1.85E+09</td>
<td>8.10E-09</td>
<td>W</td>
<td>1.50E+01</td>
<td>0.05%</td>
</tr>
<tr>
<td>Eu-152</td>
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<td>5.97E-08</td>
<td>W</td>
<td>6.09E+00</td>
<td>0.02%</td>
</tr>
<tr>
<td>Eu-154</td>
<td>2.11E-03</td>
<td>7.82E+07</td>
<td>7.73E-08</td>
<td>W</td>
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<td>0.02%</td>
</tr>
<tr>
<td>Eu-155</td>
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<td>1.12E-08</td>
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<td>0.02%</td>
</tr>
<tr>
<td>Gd-153</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>6.43E-09</td>
<td>D</td>
<td>0.00E-00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Subtotal:</td>
<td>9.15E+00</td>
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<td>9.83E-07</td>
<td></td>
<td>5.90E+03</td>
<td>20.96%</td>
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</table>

<table>
<thead>
<tr>
<th>Actinides</th>
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</tr>
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<tbody>
<tr>
<td>U-234</td>
<td>2.35E-03</td>
<td>8.70E+07</td>
<td>2.13E-06</td>
<td>W</td>
<td>1.85E+02</td>
<td>0.66%</td>
</tr>
<tr>
<td>U-235</td>
<td>7.97E-05</td>
<td>2.95E+06</td>
<td>1.97E-06</td>
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<td>0.02%</td>
</tr>
<tr>
<td>U-236</td>
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<td>0.00E+00</td>
<td>2.01E-06</td>
<td>W</td>
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<td>0.00%</td>
</tr>
<tr>
<td>U-238</td>
<td>5.17E-05</td>
<td>1.91E+06</td>
<td>1.91E-06</td>
<td>W</td>
<td>3.66E+00</td>
<td>0.01%</td>
</tr>
<tr>
<td>Np-237</td>
<td>1.77E-06</td>
<td>6.55E+04</td>
<td>1.46E-04</td>
<td>W</td>
<td>9.56E+00</td>
<td>0.03%</td>
</tr>
<tr>
<td>Pu-238</td>
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<td>1.17E+07</td>
<td>1.06E-04</td>
<td>W</td>
<td>1.24E+03</td>
<td>4.39%</td>
</tr>
<tr>
<td>Pu-239</td>
<td>4.46E-03</td>
<td>1.65E+08</td>
<td>1.16E-04</td>
<td>W</td>
<td>1.91E+04</td>
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<tr>
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<td>W</td>
<td>1.23E+03</td>
<td>4.37%</td>
</tr>
<tr>
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<td>2.67E-03</td>
<td>9.88E+07</td>
<td>2.23E-06</td>
<td>W</td>
<td>2.20E+02</td>
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</tr>
<tr>
<td>Pu-242</td>
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<td>0.00E+00</td>
<td>1.11E-04</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Am-241</td>
<td>5.75E-05</td>
<td>2.13E+06</td>
<td>2.20E-04</td>
<td>W</td>
<td>2.55E+02</td>
<td>0.91%</td>
</tr>
<tr>
<td>Am-242</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>1.58E-08</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Am-242m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>1.15E-04</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Am-243</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>1.19E-04</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Cm-242</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>4.67E-06</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Cm-244</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>6.70E-05</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Subtotal:</td>
<td>1.03E-02</td>
<td>3.80E+08</td>
<td>1.03E-03</td>
<td></td>
<td>2.23E+04</td>
<td>79.04%</td>
</tr>
</tbody>
</table>

Total: 2.82E+04 (Sv/Total)

\(^{(2)}\) Internal dose factor for tritium was increased by 50% to include skin absorption.
\(^{(3)}\) The unit dose is the product of the normalized activity and the inhalation dose factor. To convert Sv/g to rem/g, multiply by 100.
\(^{(4)}\) Kr-85 is a noble gas. It does not accumulate in the body; therefore, its internal dose factor is zero. The value shown is the external dose rate factor for submersion in an infinite cloud divided by the light activity breathing rate.

DP = daughter products (included with parents and not tracked individually).
Table D3-8. Light Water Reactor Radionuclide Composition of Fuel and the Unit Dose Factor per Assembly. (2 sheets)

<table>
<thead>
<tr>
<th>Fission and Activation Products</th>
<th>Activity (Ci/Asmbl)</th>
<th>Activity (Bq/Asmbl)</th>
<th>Inhalation dose factor (Sv/Bq)</th>
<th>Class</th>
<th>Unit dose (Sv/Asmbl)</th>
<th>Percent of Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>H-3</td>
<td>9.28E+01</td>
<td>3.43E+12</td>
<td>2.60E-11</td>
<td>Vapor</td>
<td>8.91E+01</td>
<td>0.00%</td>
</tr>
<tr>
<td>C-14</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>5.64E-10</td>
<td>Organic</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Fe-55</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>7.26E-10</td>
<td>D</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Co-60</td>
<td>7.88E+02</td>
<td>2.92E+13</td>
<td>8.94E-09</td>
<td>W</td>
<td>2.61E+05</td>
<td>0.00%</td>
</tr>
<tr>
<td>Ni-59</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>7.31E-10</td>
<td>Vapor</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Ni-63</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>1.70E-09</td>
<td>Vapor</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Se-79</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>2.66E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Kr-85</td>
<td>1.31E+03</td>
<td>4.85E+13</td>
<td>3.57E-13</td>
<td>Gas(^4)</td>
<td>1.73E+01</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sr-90</td>
<td>1.98E+04</td>
<td>7.33E+14</td>
<td>6.47E-08</td>
<td>D</td>
<td>4.74E+07</td>
<td>0.29%</td>
</tr>
<tr>
<td>Y-90</td>
<td>1.98E+04</td>
<td>7.33E+14</td>
<td>2.13E-09</td>
<td>W</td>
<td>1.56E+06</td>
<td>0.01%</td>
</tr>
<tr>
<td>Zr-93</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>8.67E-08</td>
<td>D</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Nb-93m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>8.68E-10</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Tc-99</td>
<td>5.21E+00</td>
<td>1.93E+11</td>
<td>2.25E-09</td>
<td>W</td>
<td>4.34E+02</td>
<td>0.00%</td>
</tr>
<tr>
<td>Ru-106</td>
<td>8.43E+00</td>
<td>3.12E+11</td>
<td>3.18E-08</td>
<td>W</td>
<td>9.92E+03</td>
<td>0.00%</td>
</tr>
<tr>
<td>Ru-106</td>
<td>8.43E+00</td>
<td>3.12E+11</td>
<td>DP</td>
<td></td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Pd-107</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>2.19E-10</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Ag-110</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>DP</td>
<td></td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Ag-110m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>8.34E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Cd-113m</td>
<td>1.31E+01</td>
<td>4.85E+11</td>
<td>4.13E-07</td>
<td>D</td>
<td>2.00E+05</td>
<td>0.00%</td>
</tr>
<tr>
<td>In-113m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>1.11E-11</td>
<td>D</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sn-113</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>2.88E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sn-119m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>1.69E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sn-121m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>3.11E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sn-123</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>8.79E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sn-126</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>2.69E-08</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sb-125</td>
<td>1.30E+02</td>
<td>4.81E+12</td>
<td>3.30E-09</td>
<td>W</td>
<td>1.59E+04</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sb-126</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>3.17E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sb-126m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>9.17E-12</td>
<td>D</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Te-123m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>2.86E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Te-125m</td>
<td>3.17E+01</td>
<td>1.17E+12</td>
<td>1.97E-09</td>
<td>W</td>
<td>2.31E+03</td>
<td>0.00%</td>
</tr>
<tr>
<td>Te-127</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>8.60E-11</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Te-127m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>5.81E-09</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>I-129</td>
<td>1.26E-02</td>
<td>4.66E+08</td>
<td>4.69E-08</td>
<td>D</td>
<td>2.19E+01</td>
<td>0.00%</td>
</tr>
<tr>
<td>Cs-134</td>
<td>4.91E+02</td>
<td>1.82E+13</td>
<td>1.25E-08</td>
<td>D</td>
<td>2.27E+05</td>
<td>0.00%</td>
</tr>
<tr>
<td>Cs-135</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>1.23E-09</td>
<td>D</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Cs-137</td>
<td>2.85E+04</td>
<td>1.05E+15</td>
<td>8.63E-09</td>
<td>D</td>
<td>9.10E+06</td>
<td>0.06%</td>
</tr>
<tr>
<td>Ba-137m</td>
<td>2.70E+04</td>
<td>9.98E+14</td>
<td>DP</td>
<td></td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Ce-144</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>5.84E-08</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
</tbody>
</table>
Table D3-8. Light Water Reactor Radionuclide Composition of Fuel and the Unit Dose Factor per Assembly. (2 sheets)

<table>
<thead>
<tr>
<th>Fission and Activation Products</th>
<th>Activity (Ci/Asmbly)</th>
<th>Activity (Bq/Asmbly)</th>
<th>Inhalation dose factor (Sv/Bq)</th>
<th>Class</th>
<th>Unit dose (Sv/Asmbly)</th>
<th>Percent of Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pr-144</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>1.16E-11</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Pr-144m</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>DP</td>
<td></td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Pm-147</td>
<td>8.17E+02</td>
<td>3.02E+13</td>
<td>6.97E-09</td>
<td>W</td>
<td>2.11E+05</td>
<td>0.00%</td>
</tr>
<tr>
<td>Sm-151</td>
<td>1.23E+02</td>
<td>4.55E+12</td>
<td>8.10E-09</td>
<td>W</td>
<td>3.69E+04</td>
<td>0.00%</td>
</tr>
<tr>
<td>Eu-152</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>5.97E-08</td>
<td>W</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Eu-154</td>
<td>1.55E+03</td>
<td>5.74E+13</td>
<td>7.73E-08</td>
<td>W</td>
<td>4.43E+06</td>
<td>0.03%</td>
</tr>
<tr>
<td>Eu-155</td>
<td>4.02E+02</td>
<td>1.49E+13</td>
<td>1.12E-08</td>
<td>W</td>
<td>1.67E+05</td>
<td>0.00%</td>
</tr>
<tr>
<td>Gd-153</td>
<td>0.00E+00</td>
<td>0.00E+00</td>
<td>6.43E-09</td>
<td>D</td>
<td>0.00E+00</td>
<td>0.00%</td>
</tr>
<tr>
<td>Subtotal:</td>
<td>1.01E+05</td>
<td>3.73E+15</td>
<td>9.83E-07</td>
<td></td>
<td>6.36E+07</td>
<td>0.39%</td>
</tr>
</tbody>
</table>

**Actinides**

| U-234                          | 4.72E-01             | 1.75E+10             | 2.13E-06                      | W     | 3.72E+04              | 0.00%           |
| U-235                          | 6.91E-03             | 2.56E+08             | 1.97E-06                      | W     | 5.04E+02              | 0.00%           |
| U-236                          | 1.07E-01             | 3.96E+09             | 2.01E-06                      | W     | 7.96E+03              | 0.00%           |
| U-238                          | 1.30E-01             | 4.81E+09             | 1.91E-06                      | W     | 9.19E+03              | 0.00%           |
| Np-237                         | 1.34E-01             | 4.96E+09             | 1.46E-04                      | W     | 7.24E+05              | 0.00%           |
| Pu-238                         | 1.36E+03             | 5.03E+13             | 1.06E-04                      | W     | 5.33E+09              | 33.01%          |
| Pu-239                         | 1.25E+02             | 4.63E+12             | 1.16E-04                      | W     | 5.37E+08              | 3.32%           |
| Pu-240                         | 2.12E+02             | 7.84E+12             | 1.16E-04                      | W     | 9.10E+08              | 5.63%           |
| Pu-241                         | 2.44E+04             | 9.03E+14             | 2.23E-06                      | W     | 2.01E+09              | 12.46%          |
| Pu-242                         | 9.39E-01             | 3.47E+10             | 1.11E-04                      | W     | 3.86E+06              | 0.02%           |
| Am-241                         | 9.44E-02             | 3.49E+13             | 1.20E-04                      | W     | 4.19E+09              | 25.94%          |
| Am-242                         | 1.06E+01             | 3.94E+11             | 1.58E-08                      | W     | 6.23E+03              | 0.00%           |
| Am-242m                        | 1.07E+01             | 3.96E+11             | 1.15E-04                      | W     | 4.55E+07              | 0.28%           |
| Am-243                         | 1.28E+01             | 4.74E+11             | 1.19E-04                      | W     | 5.64E+07              | 0.35%           |
| Cm-242                         | 8.82E+00             | 3.26E+11             | 4.67E-06                      | W     | 1.52E+06              | 0.01%           |
| Cm-244                         | 1.21E+03             | 4.48E+13             | 6.70E-05                      | W     | 3.00E+09              | 18.57%          |
| Subtotal:                      | 2.83E+04             | 1.05E+15             | 1.03E-03                      |       | 1.61E+10              | 99.61%          |

Total: 1.62E+10 (Sv/Asmbly)

(2) Internal dose factor for tritium was increased by 50% to include skin absorption.
(3) The unit dose is the product of the normalized activity and the inhalation dose factor. To convert Sv/g to rem/g, multiply by 100.
(4) Kr-85 is a noble gas. It does not accumulate in the body; therefore, its internal dose factor is zero. The value shown is the external dose rate factor for submersion in an infinite cloud divided by the light activity breathing rate.

DP = daughter products (included with parents and not tracked individually).
The atmospheric transport factor \( \left( \chi/Q \right) \) is based on specific release conditions (e.g., ground level or elevated, long or short duration) and the receptor's distance from the release. While the methodology is common to the SNF Project, the atmospheric transport factor is the time-integrated normalized air concentration at the receptor's location, which is a measured distance from the Canister Storage Building. The transport factor includes the dilution of an airborne contaminant caused by atmospheric mixing and turbulence. The air transport values used in this report have been generated using the GXQ computer program (WHC-SD-GN-SWD-30002 and WHC-SD-GN-SWD-30003). Table D3-9 contains the air transport values used to determine on-site and off-site consequences.

Air transport factors were calculated using methods found in NRC Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*. In each wind direction, the observed frequencies of particular wind speed and stability class combinations are used to compute a value that is exceeded only 0.5% of the time. This is repeated for all 16 compass directions to determine the worst-case location.

Exposures to the collocated worker on-site are calculated for the individual at the 100-m location. The risk evaluation guidelines apply to this individual. For assessment purposes, DOE has directed (Sellers 1996) that the Hanford Site boundary be considered the location of the off-site receptor. Highway 240 is not used as the nearest public access because DOE and its contractors can control access during emergency and accident conditions. This access control meets the requirements of 10 CFR 72.106(c).

Table D3-9. Atmospheric Transport Factors Used in Accident Analyses for the Interim Storage Area.

<table>
<thead>
<tr>
<th>Receptor location description</th>
<th>Air transport factors (^{(1)})</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Acute ( \chi/Q ) Less than 1 hour (^{(2)})</td>
</tr>
<tr>
<td>On-site worker (100 m east)</td>
<td>3.41 E-02</td>
</tr>
<tr>
<td>Highway 240 (9,280 m west)</td>
<td>2.01 E-05</td>
</tr>
<tr>
<td>Hanford Site boundary (17,390 m east)</td>
<td>1.30 E-05</td>
</tr>
</tbody>
</table>

\(^{(1)}\) Units for these values are seconds per cubic meter. In all cases, the releases are assumed to be point sources at ground level to maximize the dose consequences.

None of the accidents analyzed in this document adjusts the air transport factors for the finite size of the source (i.e., cask wake effects) or for the elevation of the release above ground level (i.e., stack effects). Section 1.4.1 of the SNF Project FSAR provides additional information on the calculation of the air transport factors. The basis for defining the location of the on-site and off-site receptors is provided in Section D1.3.1.3.

The breathing rate (BR) depends on individual activity factors and exposure duration. This methodology is common to the SNF Project facilities and is described in Chapter 3.0 of the SNF Project FSAR.

The dose per unit intake (UD) is the 50-year dose commitment for all relevant exposure pathways per gram of radioactive material inhaled. The major radiation exposure pathway for the identified accidents is inhalation of radioactive material. This methodology is common to the SNF Project facilities and is described in Chapter 3.0 of the SNF Project FSAR.

Hypothetical Release Fractions

An unmitigated release from each fuel type cask is evaluated to determine the appropriate safety classification of the engineered barriers that prevent an uncontrolled release. The hypothetical accident scenario developed in this analysis assumes a major failure of the cask with crushing of some fuel pellets. Some credit is taken for the physical presence of the cask around the fuel, even though its design confinement integrity has been breached. The hypothetical unmitigated release is developed based on the quantity of radioactive material that might be released if the fuel assembly or assemblies within the storage cask are damaged when the cask is breached. Although this scenario is hypothetical and non-mechanistic, the scenario establishes the magnitude of a potential release. Based on the postulated consequences of the unmitigated release, appropriate safety classification of the protection SSCs can then be accomplished.

For hypothetical purposes, a scenario is proposed involving a crane hook structure, with a weight of 2 tons falling from a height of 80 ft (but hypothetically retaining significant energy to crush the fuel after puncturing the cask sidewall). The tapered shape of the hook bottom is assumed to be about 2 in. across the leading edge, tapering out to about 5-in. thick at the throat of the hook. This hook creates a strong rigid projectile that (for this scenario) is hypothetically proposed to have enough energy to penetrate the cask sidewalls and shear all fuel rods contained within the cask and inner canister. This hypothetical accident results in clad damage to all of the elements and pulverization of some of the fuel that was directly struck by the crane hook. This accident would conservatively damage about a 7-in long cross-section of the fuel assembly. In this scenario, it only has sufficient energy remaining to crush half of the material damaged, which equates to about 2.5% of the fuel inventory in the cask. The effective impact energy density calculated for the hook structure falling from a height of 80 ft is 1.7E9 ergs/cm^3. Figure 4.11 of NUREG-1320 indicates that the mass fraction of particulate material less than
10 μm generated from this energy density is about 1.0% for ceramic fuel. Figure 4.15 (NUREG-1320) indicates that about 10% of the less than 10 μm particles are less than 3 μm, which is considered respirable for a particle with a density like uranium oxide. Credit is taken for the damaged cask structure retaining 90% of the particulate generated.

The parameters relating to fuel damage and release fractions for the postulated scenarios are listed below and in Table D3-10.

- Fraction of fuel pins damaged: 2.5%
- Gap noble gases, iodine, tritium: 100% released from damaged pins
- Gap particulates released: 1% released from damaged pins
- Particulates from crushed pins: 1%
- Respirable fraction of particulates: 10%
- Fraction of noble gases, iodine, tritium released from crushed pins: 100%
- Fraction of noble gases, iodine, tritium released from cask: 100%
- Fraction of particulates released from casks: 10%

The release is modeled to enter the environment as a ground level airborne release with a 1-hour duration. Radiological doses resulting from each of the postulated accidents were calculated for the maximum on-site worker and off-site individual. The analysis shown in the tables indicates that even for the extremely conservative postulated accident of an unmitigated release, radiation exposure to maximum off-site individuals would be less than 0.5 rem.

D3.4.1.3 Frequency Estimates. DOE Order 6430.1A requires that safety-class SSCs be identified if they are required to place or maintain an operating process in a safe condition when that process prevents or mitigates consequences to the public greater than 500 mrem (5 mSv) total effective dose equivalent, independent of the estimated event frequency. Therefore, the determination of frequency estimates is based on methodology common to the SNF Project facilities and is described in Chapter 3.0 of the SNF Project FSAR.

D3.4.1.4 Risk Guidelines. The DOE-recommended radiological risk evaluation guidelines (Sellers 1997) are applied across the SNF Project and are described in Chapter 3.0 of the SNF Project FSAR.

D3.4.1.5 Safety Structures, Systems, and Components. Safety-class and safety-significant designations, as related to the SSCs, are defined consistently for the SNF Project in Chapter 3.0 of the SNF Project FSAR.
Table D3-10. Dose Calculation Summary for the Postulated Releases.

<table>
<thead>
<tr>
<th>Receptor identity</th>
<th>Air transport factor, s/m³</th>
<th>Sv released</th>
<th>Rate, m³/s (1)</th>
<th>EDE, rem</th>
<th>Anticipated evaluation guidelines (2)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Fast Flux Test Facility Fuel</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>On-site worker, 100 m east-southeast</td>
<td>3.41E-02</td>
<td>1.10E+05</td>
<td>3.33E-04</td>
<td>1.25E-02</td>
<td>1</td>
</tr>
<tr>
<td>Highway 240, 9,280 m west</td>
<td>2.36E-05</td>
<td>1.10E+05</td>
<td>3.33E-04</td>
<td>8.63E-02</td>
<td>0.5</td>
</tr>
<tr>
<td>Hanford boundary, 17,390 m east</td>
<td>1.30E-05</td>
<td>1.10E+05</td>
<td>3.33E-04</td>
<td>4.75E-02</td>
<td>0.5</td>
</tr>
<tr>
<td><strong>NRF TRIGA Fuel</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>On-site worker, 100 m east-southeast</td>
<td>3.41E-02</td>
<td>7.85E-02</td>
<td>3.33E-04</td>
<td>9.01E-03</td>
<td>1</td>
</tr>
<tr>
<td>Highway 240, 9,280 m west</td>
<td>2.36E-05</td>
<td>7.85E-02</td>
<td>3.33E-04</td>
<td>6.24E-06</td>
<td>0.5</td>
</tr>
<tr>
<td>Hanford boundary, 17,390 m east</td>
<td>1.30E-05</td>
<td>7.85E-02</td>
<td>3.33E-04</td>
<td>3.44E-06</td>
<td>0.5</td>
</tr>
<tr>
<td><strong>Commercial Light Water Reactor Fuel</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>On-site worker, 100 m east-southeast</td>
<td>3.41E-02</td>
<td>4.05E+04</td>
<td>3.33E-04</td>
<td>4.60E-01</td>
<td>1</td>
</tr>
<tr>
<td>Highway 240, 9,280 m west</td>
<td>2.36E-05</td>
<td>4.05E+04</td>
<td>3.33E-04</td>
<td>3.19E-02</td>
<td>0.5</td>
</tr>
<tr>
<td>Hanford boundary, 17,390 m east</td>
<td>1.30E-05</td>
<td>4.05E+04</td>
<td>3.33E-04</td>
<td>1.76E-02</td>
<td>0.5</td>
</tr>
</tbody>
</table>

(1) The average breathing rate is: light activity breathing rate.
(2) Guideline is for the "anticipated" accident category.

EDE = effective dose equivalent.

**D3.4.2 Design Basis Accident Analysis**

Although the 200 Area ISA is a new facility, the development of safety documentation for a new project per HNF-PRO-703, Safety Analysis Process - New Project, is different for this FSAR. The safety bases for the FFTF SNF and the TRIGA SNF storage systems and the 400 Area ISA site were approved under a DOE Richland Operations Office-approved safety
authorization basis at FFTF (WHC-TI-75002, Appendix H, Amendment 77 and Wagoner 1996). These storage systems and SNF types have been analyzed and approved for storage at the 400 Area ISA. Although these analyses were performed for the 400 Area ISA, information relevant to the performance of the FFTF and TRIGA storage systems at the 200 Area ISA can be derived from many of the analyses.

The existing analyses for the FFTF, TRIGA, and LWR spent fuel were evaluated for their applicability to the 200 Area ISA. The evaluations reviewed the assumptions, methodology, parameters, parameter values, results, and conclusions of the 400 Area ISA analyses for application to the 200 Area ISA. The results of these evaluations are provided in this section. The evaluations generally found that the analyses that relied solely on the performance and functional characteristics of the storage systems could be applied from the 400 Area ISA to the 200 Area ISA, but where site characteristics of the 400 Area ISA were not enveloped by the site characteristics of the 200 Area ISA, new analyses were performed.

The results of these evaluations and the 200 Area ISA site-specific analyses provide the bases to: (1) identify the hazards, energy sources, potential accident sequences, available mitigating barriers and controls, and qualitative estimates of accident frequency and consequence; (2) evaluate the most significant accident(s) to provide the technical bases for the conclusions related to safety, health, and the environment (e.g., final safety designation and comparison with risk guidelines); and (3) provide a final assessment of the design relative to maintaining its confinement function under normal, abnormal, and accident conditions.

This section presents a summary of the key assumptions and results of the DBA analyses that have been performed for the ISA. The DBAs are summarized based on the guidelines provided in DOE-STD-3009-94, and include the following categories:

- Cask handling/drop
- Mobile crane fall
- Cask tipover
- Fuel rod rupture
- Seismic
- Tornado/wind
- Fire.

The DBAs have been analyzed to quantify consequences and compare them with release limits for off-site consequences and evaluation guidelines for on-site consequences. The process is iterative, starting by taking no credit for mitigative features and comparing the results to the limits or guidelines. Credit is then taken for safety SSCs that prevent or mitigate the consequences to show that the results are below the release limits and evaluation guidelines. The process continues after the release limits and evaluation guidelines are met by identifying other SSCs that, while not designated as Safety Class or Safety Significant, provide additional mitigative features as defense in depth.
D3.4.2.1 Handling and Drop Accidents.

D3.4.2.1.1 Scenario Development. While handling the storage casks, an accidental drop or other handling impact could possibly occur. If the drop or impact caused a cask to undergo accelerations and forces beyond the design strength, the cask confinement could be breached. Breach of the cask confinement could result in radiological releases. In addition, as a passive design feature, each FFTF ISC and Nuclear Assurance Corporation (NAC)-1 cask protects the geometry of the stored SNF during and after impact to assure criticality safety. The handling and drop scenarios are developed in this section for each fuel type. A summary of the safety features required to prevent handling and drop accidents is provided in Table D3-11.

Fast Flux Test Facility Fuel

The FFTF fuel package consists of fuel assemblies or pin containers contained within a core component container (CCC), which in turn is contained within an ISC. One of the CCC drop analyses consists of a postulated drop into the ISC. Although the scope of this analysis does not include CCC removal from or placement in the ISC, the analysis provides a measure of the impact strength of the CCC to ensure that the CCC can remain intact inside the ISC for geometry (criticality) control. The ISC, with payload, has also been analyzed to ensure that confinement of the fuel is maintained.

The CCC was analyzed (WHC-SD-FF-DA-077) to confirm that it can safely withstand accident conditions. Fully loaded CCCs (six or seven fuel assemblies, or six pin containers) are addressed in WHC-SD-FF-DA-077, Stress and Structural Analysis of the Core Component Container (CCC), and SNF-4790, 200 Area ISA Design Basis Accident Analysis Documentation for FFTF Fuel. Partially loaded CCCs are not analyzed, and must be evaluated on a case-by-case basis. Material properties were evaluated using a peak CCC temperature condition of 600 °F. Normal CCC conditions were evaluated in accordance with stress limits of the ASME Code, Section VIII, Division 2 (ASME 1989), and the accident conditions were evaluated in accordance with stress limits of the ASME Code, Section III, Appendix F (ASME 1989).

The accident conditions analyzed for the CCC are as follows:

- 18-ft drop of the CCC into the ISC
- 4-ft sideways drop of the CCC while inside the ISC.

Several drop analyses have been performed for the ISC. Under normal operations, the ISC is lifted vertically by attaching the lifting fixture to three anchor attachments imbedded in the concrete of the ISC. No side lifting is allowed. In accordance with the critical lift requirements of DOE/RL-92-36, Hanford Site Hoisting and Rigging Manual, the ISC lift points are designed to lift five times the weight of the cask, without exceeding the ultimate stress of the material, or three times the weight, without exceeding the yield strength of the material, whichever is less. Therefore, the normal ISC lifting and handling loads generate low stresses.
Table D3-11. Summary of Safety Features Required to Prevent Handling/Drop Accident.

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator(s)</th>
<th>Safety function</th>
<th>Safety features (described in Chapter D4.0)</th>
</tr>
</thead>
</table>
| Drop of the cask while being moved by a crane | G-03, G-06, G-13 | Maintain geometry control after credible drops. Maintain confinement of radioactive materials after a credible drop and provide passive protection such that structural integrity is maintained. | Safety class equipment for criticality geometry control:  
- CCC  
- LWR canister.  
Safety-significant equipment for confinement:  
- FFTF ISC  
- NRF TRIGA cask  
- 2R container  
- LWR canister.  
Safety-significant equipment for structural integrity:  
- DOT-6M container  
- FFTF ISC  
- NAC-I cask.  
TSR:  
- Restriction on cask lift heights.  
- Restriction on lifting casks over objects.  
Defense-in-depth:  
- Qualified crane operators  
- Detailed procedures. |

(1) Checklist designators are from SNF-4820, 1999, 200 Area Interim Storage Area Final Hazard Analysis, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.

2R = inner container of DOT-6M.
CCC = core component container.
DOT = U.S. Department of Transportation.
FFTF = Fast Flux Test Facility.
ISC = interim storage cask.
LWR = light water reactor.
NAC = Nuclear Assurance Corporation.
NRF = Neutron Radiography Facility.
TRIGA = Training, Research and Isotope Production, General Atomics.
TSR = Technical Safety Requirement.
The weakest link in the ISC attachment design is the anchor lug, which screws into the anchor bolt of the cask. If a lifting accident occurs whereby a lift point fails, this anchor will fail before the ISC concrete embedment. Thus, the ISC integrity or shielding will not be affected.

The accident conditions analyzed for the ISC are as follows:

- 18-ft drop of the CCC into the ISC
- 4-ft sideways drop of the ISC
- 4-ft ISC drop onto an unyielding surface
- 8-ft ISC drop with the CCC inside onto a 1.5-ft thick concrete pad
- ISC drop onto another object
- 40-in. ISC drop onto a 6-in. diameter mild steel punch.

**18-ft drop of the CCC into the ISC.** The ISC internal impact limiter is used to absorb the impact of a fully loaded CCC from a height of 18 ft. The impact limiter consists of a 1.5-in. layer of aluminum honeycomb, with a compressive strength of 4,400 lbs/in², and a CCC impact area of 153 in².

The dynamic finite-element analysis (WHC-SD-FF-DA-077) shows that the CCC will remain intact during and after the 18-ft vertical drop into the ISC, and the CCC boundary will provide adequate sealing during and after the drop. The results of the stress evaluation indicate that the stresses in some of the components exceed the yield strength but they are within the stress limits of the ASME Code, Section III, Appendix F (ASME 1989). The magnitude of the CCC tube plastic deformation does not affect the integrity of the CCC or the seals. The critical component of the CCC for membrane stress is the central tube of the CCC. The maximum membrane compressive elastic stress is 37,313 lbs/in², which is less than the allowable 39,360 lbs/in².

If an assembly is stored in the center tube of the CCC, it will rebound and hit the CCC cover and the bolts, and the seal region will remain elastic. The impact limiter over the central tube limits the load to the socket region of the CCC cover to 8,000 lbf. The metal O-ring seal of the CCC cover will have a maximum opening of 0.005 in., which is within the manufacturer's deformation limit to retain sealing effectiveness. The seal will neither lose contact with the cover nor with the upper support flange. The CCC cover bolts are subject to a tensile stress of 61,666 lbs/in², which is less than the allowable 119,000 lbs/in².

**4-ft sideways drop of the CCC while inside the ISC.** The CCC, inside the ISC, is subjected to a 96.1-g maximum side loading during a 4-ft drop (General Atomics 1995). The ISC remains leaktight during this event. If it is assumed that the CCC rests on the ISC wall, the results show that the CCC material will not exceed the yield strength of the materials. Thus, no permanent deformation will occur.
If it is assumed that the CCC has 0.5 in. that can be deflected within the ISC liner, the dynamic finite-element analysis case shows that the elastic stresses in some of the components exceed the yield strength, but they are within the stress limits of the ASME Code, Appendix F, Section III, Division 1 (ASME 1989). The conditions of this analysis assume that the fuel element within the CCC hits the tube wall, and then the tube wall and element move together and hit the side of the ISC. The stress results show the bending stress in the storage tube is 51,608 lbs/in², which is less than the membrane-plus-bending stress allowable of 59,040 lbs/in². The magnitude of the plastic deformation is not expected to affect the integrity of the CCC because stresses are within allowable limits. Note that a 0.5-in. gap is not expected because the CCC will rest on the wall of the ISC during the drop.

4-ft ISC drop onto an unyielding surface. The 4-ft drop analysis consists of two separate drops: an end-drop and an angle drop with slap down. The maximum stresses on the ISC confinement barrier due to the 4-ft end-drop condition onto an unyielding surface occur in the 1.5-in. thick inner cylinder. A shell/ring discontinuity analysis was used to determine the stress at the upper flange-to-cylinder junction, and the maximum stresses were determined to be only 16.54 ksi. The g-level during a 4-ft drop onto an unyielding surface is between 70 g and 96 g. The stresses are within the normal condition allowable limits. Additionally, the bolt stress due to the flange rotation is within normal limits. Therefore, the ISC closure will not be affected during the accident conditions, and the sealing integrity is maintained.

The maximum calculated stress on the ISC confinement barrier occurs during the ISC slap down from an angled 4-ft drop. The maximum stress component is the inner 1.5-in. thick cylinder. The maximum stress that occurs for this condition is 17.4 ksi, and the maximum shear stress is 3.9 ksi. These stresses are within limits.

For the drop accidents, strength of material calculations following the methods of ACI-349, Code Requirements for Nuclear Safety Related Concrete Structures, indicate that the steel and concrete composite shielding sections of the cask will not be compromised during the drop events. The maximum amount of concrete crush during this accident is 6 in.

8-ft ISC drop onto a 1.5-ft thick concrete pad. The 4-ft drop analyses were conducted assuming a hard, unyielding surface. However, the ISA storage pad is a 1.5-ft thick concrete pad that will absorb much of the impact energy. To ascertain the effects of a yielding surface, an analysis was also performed for an 8-ft drop onto a 1.5-ft thick concrete pad. The highest g-level is 46 g. Analyses of these drops (SNF-4790) show that they are bounded by the 4-ft drop onto an unyielding surface.

Based on the drop scenarios postulated, confinement and geometry control are not lost for an 8-ft drop of the ISC package (including the CCC and fuel payload) onto the 200 Area ISA storage pad.
**ISC drop onto another object.** The ISC could be moved over other objects during handling. If the ISC drops onto an object, the ISC could strike the object and then tip over, resulting in a horizontal drop onto the 200 Area ISA storage pad. If the object struck by the ISC is taller than 4 ft, the horizontal drop will be outside of the bounds of the drop analyses. Therefore, the ISC is assumed to not be moved over objects that are taller than 4 ft. Note that the cask trailer is less than 4 ft tall.

**40-in. ISC drop onto a 6-in. diameter mild steel punch.** The ISC was evaluated for the hypothetical accident condition 40-in. drop of the ISC onto a 6-in. diameter mild steel punch. Conservatively, 1.3 times the weight for casks with diameters less than 30 in. was used. Only the concrete is used to stop the punch, even though there is enough steel to stop the punch also.

It was concluded that the ISC is of sufficient thickness to preclude punching shear failure. The primary stresses across the cask cross-section and the closure and bottom plates due to a puncture drop event are always less than 26.2 ksi, which are less than the allowable values of 63.3 ksi for primary membrane plus bending stress. The stainless steel closure, and bottom plate, and the cask body sidewall are of sufficient thickness to preclude punching sheer failure.

**NRF TRIGA Fuel**

The NRF TRIGA fuel could arrive at the ISA storage pad in either casks or U.S. Department of Transportation (DOT)-6M containers, which are then placed into the Rad-Vault storage container for long-term storage. During handling operations, an NRF TRIGA cask, DOT-6M container, or Rad-Vault could be involved in a drop accident. Handling and drop analyses were performed to analyze the effects of drops. The following are the types of drops that could result in significant impact to one of the NRF TRIGA packages:

- Cask drop onto another cask or Rad-Vault
- Rad-Vault lid drop
- NRF TRIGA cask drop
- DOT-6M container drop.

**Cask drop onto another cask or Rad-Vault.** The cask drop onto another cask or Rad-Vault is bounded by the crane boom impact analysis discussed in Section D3.4.2.2, and is not addressed further in this section.

**Rad-Vault lid drop.** The Rad-Vault is not lifted while loaded with contents or with the lid in place (CNSI 1992). The Rad-Vault lid will be placed over the body after the cask or DOT-6M container is placed inside. The lid of the Rad-Vault, if dropped, will not drop onto the cask or DOT-6M containers stored in the Rad-Vault because the cylindrical design physically precludes this event (the outside diameter of the lid [114-in.] is much greater than the inside diameter [79-in.] of the vault body). Rad-Vault handling limits (TSR) preclude lifting the Rad-Vault cover more than 0.3 m (12 in.) above the top of the Rad-Vault in order to minimize...
potential structural damage to the Rad-Vault. Therefore, no scenario has been identified that would release radioactive material as a result of dropping the Rad-Vault lid.

**NRF TRIGA cask drop.** During handling operations, any one or all of the NRF TRIGA casks, DOT-6M containers, or the Rad-Vault could be involved in a drop accident that threatens the confinement. A handling/drop analysis was performed (WHC-SD-FF-TI-043) to analyze the effects of a credible drop. In reality, the worst-case off-normal handling scenario would be the dropping of an NRF TRIGA cask onto one or more other casks. The Rad-Vault is in place before any fuel is moved, and the DOT-6M containers are the last to be loaded into the Rad-Vault and the first to be removed. A cask-on-cask drop is bounded by the crane boom drop accident. The lid of the Rad-Vault, if dropped, will not drop onto the fuel casks or containers already stored in the Rad-Vault because the cylindrical design physically precludes this event. The analysis demonstrates that under credible drop conditions, the NRF TRIGA casks and DOT-6M containers do not release radioactive material. The 10 ft/min set down load was equated to a 1-ft drop and determined to be bounded by the 109-in. NRF TRIGA cask drop without the impact limiters and the 30-ft DOT-6M container drop with impact limiters. However, to preclude exceeding the design criteria and the analyzed conditions, lifting limits have been imposed and are identified in Section D3.4.2.1.5.

**DOT-6M container drop.** Tests were performed on DOT-6M shipping packages (with 2R inner containers) to evaluate their response to a drop from 9 m (30 ft) onto an unyielding surface, and a 1-m (40-in.) drop onto a steel punch bar. A detailed observation of the packages following the drop testing showed the bolts to be tight and no damage to other areas that would indicate the possible loss of confinement (Martin Marietta 1992).

The two DOT-6M containers will have TRIGA fuel follower control rod (FFCR) payloads instead of the standard fuel payload. The only differences between the packages tested and the TRIGA FFCR configuration are the type of seal used in the 2R inner container, and the anticipated contents within the packages. The drop-tested packages used an elastomeric O-ring seal, whereas the TRIGA FFCR shipping packages will use a metallic O-ring seal to assure that long-term storage requirements are met. In addition, the drop-tested package used can-and-bottle internal containers, whereas the TRIGA FFCR shipping packages will contain stainless steel clad fuel rods enclosed in stainless steel pipe. The TRIGA FFCRs are not externally contaminated and the solid hydride fuel itself is enclosed within the stainless steel cladding, which has been demonstrated not to leak.

The loaded DOT-6M container has been shown to survive drops at heights greater than the NRF TRIGA cask. However, to ensure that administrative controls are easily promulgated into field operations, the administrative restriction for the NRF TRIGA cask maximum lift height is conservatively applied to the DOT-6M containers (21 in. above the top of the Rad-Vault, with the lid removed).
Light Water Reactor Fuel

A 250-ton crane is used to place the NAC-1 storage casks within the associated International Standards Organization (ISO) shipping containers at the 200 Area ISA. The worst-case handling or drop scenario is for the NAC-1 cask to be dropped during cask handling or transfer operations. Administratively, only one cask/container will be handled at a time, and maximum lift heights are imposed based on this analysis.

The NAC-1 casks are transported and placed at the 200 Area ISA while inside ISO containers. The presence of the NAC-1 cask within the ISO shipping container was not considered in the NAC-1 SAR bounding cask drop scenario. However, the cask tie-down devices (cask rotation trunnions) are designed to fail under excessive loads in the trunnion tubes, such that cask integrity and shielding are not affected. During the postulated drop events, the cask mounted within the ISO shipping container provides additional energy absorption capabilities that were conservatively ignored in the accident analysis. Thus, the NAC-1 cask drop is assumed to be bounding over the ISO container drop.

The types of handling and drop impacts analyzed for the NAC-1 cask are as follows:

- Drop onto unyielding surface in various orientations
- Drop onto steel rod
- Drop onto another object.

**Drop onto unyielding surface in various orientations.** Lawrence Livermore National Laboratory document UCID-21246, *Dynamic Impact Effects on Spent Fuel Assemblies*, assessed the effects of dynamic impacts (to be expected from cask drop or tipover incidents) on the integrity of the fuel rod cladding for zircaloy-clad LWR spent fuel assemblies during cask handling and storage. The conclusions of this report define acceptable g-load limits for fuel assemblies internal to the package and state:

"The analysis of the capability of spent fuel rods to resist impact loads caused by storage cask accidents indicates that, for the most vulnerable fuel assembly, axial buckling varies from 82 g at initial storage to 95 g after 20-year storage. In a side drop, no yielding is expected below 63 g at initial storage to 74 g after 20-year storage. For storage casks designed to limit loads at or below these g levels, it is not likely that damage will occur to the spent fuel rods."

Individual fuel rods stored in the NAC-1 casks are packaged in containers that retain the cross-sectional geometry within the inner canisters. The individual rods therefore remain oriented similar to the rods in the fuel assemblies and are subject to similar buckling loads and conclusions.
The drop onto an unyielding surface also considers potential impacts to the shielding associated with the NAC-1 cask. The drop analyses (NAC 1990) consist of a postulated 30-ft drop onto an unyielding surface in the following orientations:

- Top end impact
- Top corner impact
- Side impact
- Bottom end impact
- Bottom corner impact.

**Top end impact.** The top end impact analysis focuses on the crush strength and crush area of the balsa wood in the upper impact limiter, and assumes that the balsa crush strength and crush area are constant during the impact event. The analysis also investigates the bending of the cask ring structure to ensure that the ring structure has sufficient rigidity as a base for the compressive loads of the balsa. This analysis concludes that the stress of the material in the ring structure is well below the yield strength and will provide a sufficient base (without yielding) for the impact limiter to crush against. The analysis also determined the impact deceleration loads for the top impact condition would not exceed 44.67 g, which is within the 82 g allowable.

**Bottom end impact.** The analysis for the bottom end impact assumes that the impact energy transferred to the bottom of the cask is attenuated by the lower ring structure. During normal operation, the lower ring structure is designed with enough rigidity to be used as a pedestal when the cask is in the upright configuration. However, it also functions as a sacrificial structure that deforms to dissipate impact energies. The representative analysis of the impact indicates that the bounding deceleration load is 76.6 g, which is within the allowable load of 82 g.

This calculated information was used as an input value to determine if the lead shielding (located between the inner and outer cask shell) settles or slumps, based on the magnitude of the deceleration load. The analysis determined that the 1-in. step provided in the top flange between the shells is adequate to keep the lead from slumping. This analysis neglected the resistance imposed by the copper heat transfer fins and the radial support provided by the inner and outer stainless steel shells. It was also determined that the lead will not yield in direct shear to allow the lead to slump by shear into a void region left by the original placement melt.

**Side impact.** Impact on the side of the cask is attenuated by deformation of the top and bottom ring structures, the expansion chamber, and the neutron shield chamber. Since the neutron shield chamber is empty, these four components are considered to be sacrificial. The analysis documented the energy capacity of the various components separately and plotted the values versus a common parameter. The composite energy absorption capacity of the structure was then equated to the total impact energy to arrive at the deformation impact to the sacrificial structures. From this approach, it was determined that the bottom ring structure will deform.
approximately 8.75 in. and the top ring structure will deform 7.5 in., absorbing the impact energy as designed, which results in an adequate and acceptable condition.

The ring structure was analytically checked for buckling in the side drop analysis. The analysis determined that the critical buckling stress of the inner portion of the ring structure is in excess of the dynamic compressive yield strength of the material. Therefore, compression yielding precludes buckling, and the side impact deformation analysis is adequate.

The side impact deceleration load was determined in order to evaluate whether the load imposed by the lead shielding on the inner stainless steel liner is great enough to cripple the liner. The deceleration load was determined to be 96 g on the lead between the inner and outer shells. The analysis assumed that the lead will act as a fluid. As such, it was determined that the hydrostatic pressure on the inner shell imposed by the lead will be 260 lbs/in². Analyses in the NAC-1 SAR for the contraction of molten lead around the inner shell during cask fabrication indicate that the inner shell withstands hydrostatic pressures of up to 506 lbs/in². Thus, the 260 lbs/in² imposed by the lead during the drop is within allowable pressures for the inner shell.

The side impact deceleration load was also compared against the allowable g-loads for the fuel assemblies. The 96 g deceleration load includes a peaking factor of 2.0, multiplied in the equation to calculate the force on the lead shielding and the inner shell due to the lead. The acceleration within the inner shell does not require the peaking factor, since it was used to consider local effects at the end stiffening rings, which are subject to large deformation. These deformations were calculated to be 8.75-in at the lower ring and 7.5-in at the top ring. The maximum g load is calculated as the drop height (30 ft or 360 in.) divided by the minimum crush distance (7.5 in). Therefore, \( G = \frac{360}{7.5} = 48 \) g applied to the fuel assemblies for the side drop of the NAC-1 cask. This is below the 63 g load required to cause yielding of the fuel assembly; therefore, no loss of structural integrity is expected for the side drop of the NAC-1 cask.

**Top corner impact.** This analysis investigates the loads imposed on the top of the cask structure when subjected to a top corner impact. Component calculations include the shearing resistance of the top impact limiter, the top corner impact deceleration loads, and the top corner impact bolt stress. The calculation determines that the loads imposed by the top corner impact are less than the bolt preload stress, which results in an adequate and acceptable condition.

**Bottom corner impact.** The bottom corner impact energy was analyzed assuming the maximum damage to the cask would be in the area where a drain valve is directly over the point of impact. The kinetic energy in the cask was assumed to primarily be dissipated by the major impact limiter components (i.e., the balsa, the shock tube, the gusset plates and the fin plates). Load models were developed to determine deformation loads and dissipation energies. The analysis concluded that cask kinetic energy can be absorbed by deformation of the sacrificial components external to the base of the cask that contains and surrounds the drain penetrations and minor deformation of the cask boundary. This area was considered safe during this design impact loading.
For the 30-ft drop in various orientations onto an unyielding surface, the NAC-1 SAR analyses conclude that the confinement and shielding integrity of the NAC-1 cask will be maintained, although significant damage will occur to the exterior, sacrificial structures of the cask.

**Drop onto steel rod.** This accident scenario evaluated the cask free-fall of 40-in. onto the end of a mild steel pin to ensure that no loss of confinement occurs. The analysis (NAC 1990) evaluated three locations determined to be the most likely for maximum damage to occur. These locations are as follows:

- Impact on the midspan of the cask body
- Impact of the side of the cask lid
- Direct impact on a valve.

For the impact on the midspan of the cask body, the outer shell of the cask was analyzed in relation to the 40-in. free drop. Dropping the cask such that the pin impacts the midsection of the outer shell imposes the greatest bending moment into the cask. The analysis determined that the maximum stress is small compared to the allowable stresses and that little, if any, damage to the cask will result from this drop.

The second analysis investigated the impact of the pin on the cask lid from the 40-in. drop height. The analysis determined that the cask lid bolts are not loaded in shear since the clearance between the cask lid and flange is less than the bolt hole clearance. The analysis concluded that the cask lid would be radially displaced $4.4 \times 10^{-3}$ in. at the flange seal interface. Since the NAC-1 cask is not a confinement boundary, this result has no safety consequence.

The third analysis for this section evaluated the pin loading from the 40-in. free drop of the cask on a confinement valve. The valves are recessed within the stainless steel flanges at the closure (top) end of the cask. The valves do not protrude from the cask nor are they located in trunnions. This design approach provides protection for the valves by the inherent stability of the flange sections in this hypothetical accident scenario. The analysis of the valve impact on the steel rod concluded that steel flanges will adequately protect the valves.

For the 40-in drop onto a steel rod, the NAC-1 SAR analyses concluded that the confinement and shielding integrity of the NAC-1 cask will be maintained. Only minimal damage to the cask, namely the localized deformation and slight puncture of the shield tank, will be incurred.

**Drop onto another object.** The analysis for the drop of the NAC-1 cask does not consider impact onto another object and subsequent potential slapdown. Therefore, it is assumed that the NAC-1 casks will not be lifted over other objects except the transport trailer.
D3.4.2.1.2 Source Term Development. For handling and drop accidents, there is no source term from any of the fuel packages for the events postulated and analyzed in the scenario development for any of the package types. However, the events and analysis depend on initial conditions that could be violated during operations. For example, the ISC has been shown to survive an 8-ft drop onto the 200 Area ISA pad, but a drop from a higher distance will represent an unanalyzed condition. If a drop occurs from a distance higher than those analyzed, catastrophic damage is not expected because of the margin and conservatisms in the analyses. But since the package responses are unknown, a drop from heights greater than those postulated in the scenario development are bounded by the source terms calculated for the worst-case, non-mechanistic release shown in Section D3.4.1 for each fuel type.

D3.4.2.1.3 Consequence Analysis. All of the package types have been shown to maintain confinement of the fuel for the handling and drop events postulated and analyzed in the scenario development. In addition, the analyses have shown that shielding is maintained. For the events analyzed, there are no on-site or off-site releases of hazardous materials and no direct radiation hazards to workers greater than those expected during normal operations. A consequence analysis is not needed for the FFTF, TRIGA, or LWR fuel storage systems.

D3.4.2.1.4 Comparison to Guidelines

Unmitigated event – For drops that exceed the initial conditions of the analyses, the bounding consequences for each fuel type show that the off-site release limits are not challenged. Therefore, no safety-class confinement SSCs are necessary.

For the NRF TRIGA fuel, the on-site consequences do not exceed the on-site risk evaluation guidelines. Therefore, for the purposes of controlling accident releases following a handling or drop accident, no safety-significant SSCs are required to confine the NRF TRIGA fuel. However, the NRF TRIGA casks do provide structural protection of the SNF payload in the event of a drop, and as such, are designated as important-to-safety Category B for NRC equivalency. On this basis, the NRF TRIGA casks are designated as Safety Significant.

For the FFTF fuel and the LWR fuel in the NAC-I casks, the bounding on-site consequences for drops outside of the initial conditions analyzed exceed the on-site risk evaluation guidelines. Therefore, safety-significant SSCs are required to prevent or mitigate the dose consequences.

While no safety-class SSCs are required to provide confinement, some SSCs are required to protect the geometric configuration of the fuel for criticality safety. SSCs credited with retaining the critically safe geometry of the fuel are designated as Safety Class.

Mitigated event – For the handling and drop events postulated and analyzed in the scenario development, the potential releases are prevented. These events take credit for the
packaging components and limitations on handling (e.g., lift height) that have been shown to provide the impact protection and confinement of the fuel for the analyzed drop events.

**D3.4.2.1.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls.** The following SSCs and TSR controls ensure that the SNF is adequately confined and protected from inadvertent nuclear criticality during and following a drop event.

**Fast Flux Test Facility Fuel**

The following SSCs are designated as Safety Significant or Safety Class:

- **FFTF ISC** – The ISC is designated as Safety Significant to maintain confinement of radioactive materials after a credible drop and to provide passive protection of the CCC such that it retains structural integrity.

- **CCC** – The CCC is designated as Safety Class to maintain structural integrity to provide criticality geometry control.

The following TSR controls are necessary to protect the assumptions made in the drop analyses:

- **Lift height restriction** – The ISC cannot be lifted more than 8 ft from the surface of the 200 Area ISA pad.

- **Lift over object restriction** – The ISC cannot be lifted over an object that is taller than 4 ft or contains radioactive material.

**NRF TRIGA**

The following SSCs are designated as Safety Significant:

- **NRF TRIGA casks** – The TRIGA casks are designated as Safety Significant for NRC important-to-safety considerations to maintain confinement of radioactive materials after a credible drop.

- **DOT-6M container** – The DOT-6M containers are designated as Safety Significant for NRC important-to-safety considerations to provide impact absorption to maintain structural integrity for the 2R container.

- **2R container** – The 2R containers are designated as Safety Significant for NRC important-to-safety considerations to maintain confinement of radioactive materials after a credible drop.
The following TSR controls are necessary to protect the assumptions made in the drop analyses:

- **Lift restriction** – Based on the assumptions made in the drop analyses, the NRF TRIGA casks and DOT-6M containers cannot be lifted more than 109 in. (approximately 9.1 ft) from the surface of the ground at the 200 Area ISA.

- **Lift restriction** – The Rad-Vault cover shall not be lifted more than 12 in. above the top of the Rad-Vault.

**Light Water Reactor Fuel**

The following SSCs are designated as Safety Significant or Safety Class:

- **NAC-1 cask** – The NAC-1 cask is designated as Safety Significant to provide passive protection of the inner canister such that it retains structural integrity after a credible drop.

- **LWR canister** – The LWR canister is designated as Safety Significant to maintain confinement of radioactive materials after a credible drop and Safety Class to maintain structural integrity for criticality geometry control.

The following TSR controls are necessary to protect the assumptions made in the drop analyses:

- **Lift restrictions** – The NAC-1 package cannot be lifted more than 30-ft from the surface of the 200 Area ISA storage pad.

- **Lift over object restriction** – The NAC-1 package cannot be lifted over other objects except the transport trailer.

**D3.4.2.2 Mobile Crane Fall.**

**D3.4.2.2.1 Scenario Development.** While handling the storage casks, an accident could possibly occur. The casks will be picked up and transferred from current locations at the 400 Area ISA and the 300 Area, and set down at the 200 Area ISA by a substantial crane. It is likely that different mobile cranes will be used in the different areas. The FFTF crane is the Manitowoc 4000W crawler, and the 200 Area ISA crane will be a Manitowoc 250T mobile crane. The 250T Manitowoc crane is considered bounding, and analyses were performed based on the structural components of this crane.

A mechanical failure could initiate a drop of a crane boom onto the SNF storage systems. If the impact caused forces beyond the design strength, the cask containment could be breached.
Breach of the cask containment could result in radiological releases. The casks also protect the inner canister, which maintains the geometry of stored SNF to meet any criticality safety concerns. The crane boom was demonstrated to buckle under the lateral impact load before it can impart sufficient energy to breach the cask systems (SNF-4794). A summary of the safety features required to prevent the mobile crane fall accident is provided in Table D3-12.

Table D3-12. Summary of Safety Features Required to Prevent Mobile Crane Fall Accident.

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator (1)</th>
<th>Safety function</th>
<th>Safety features (described in Chapter D4.0)</th>
</tr>
</thead>
</table>
| Drop of a boom onto the cask | G-03 | Maintain confinement of radioactive materials after the crane fall. | Safety-significant equipment for confinement: 
  • FFTF ISC.  
Safety-significant equipment for structural integrity: 
  • NAC-I cask system  
  • FFTF ISC.  
TSR:  
  • Restriction on use of crane and lifting equipment. 
Defense-in-depth:  
  • Qualified crane operators  
  • Detailed procedures. |

(1) Checklist designators are from SNF-4820, 1999, 200 Area Interim Storage Area Final Hazard Analysis, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.

FFTF = Fast Flux Test Facility.  
ISC = interim storage cask.  
NAC = Nuclear Assurance Corporation.  
TSR = Technical Safety Requirement.
Fast Flux Test Facility

Previous 400 Area ISA analyses (WHC-SD-FF-DA-078) were performed to assess the damage that a crane boom drop onto the ISC could have on its containment function. The analysis assumed that the crane was a Manitowoc 4000W crawler and stated that the results required additional evaluation for any other crane or for the same crane at a different elevation above the top of an ISC. A supplemental analysis has been prepared for the 200 Area ISA that evaluates the effects of using the 250T Manitowoc crane at the 200 Area ISA (SNF-4794).

Conservative analysis shows significant damage to the outer concrete of the ISC from the crane boom drop or crane hook drop, but the inner steel confinement and shields will not be damaged by the event. Under these conditions, the ISC will not credibly lose its containment.

NRF TRIGA

A mobile crane fall event was assessed that assumed a mechanical failure of the crane that resulted in the boom falling onto the Rad-Vault. The Rad-Vault and NRF TRIGA cask were originally procured as general-service items, so no credit was taken for the protection they provide to the fuel assemblies. The Rad-Vault is 2.9 m (114 in.) in diameter by 2.8 m (111 in.) high. Failure of the crane would only impact a fraction of the fuel assemblies. It is not expected that the full length of any assembly would be crushed. It is assumed that the isotope activity of the entire inventory of the Rad-Vault (101 fuel elements) is equal to 101 times the isotope activity of the maximum irradiated fuel element. The assumptions used to model the unmitigated release bound this potential boom drop accident scenario and are within guidelines. No features are credited to prevent or mitigate this TRIGA fuel accident.

Light Water Reactor Fuel

An analysis was performed to determine if the impact on the NAC-1 storage cask from the postulated crane boom drop was sufficient to breach cask confinement (SNF-4794). The analysis compared the boom drop accident to the cask 30-ft drop and 40-in. puncture drop accident events that were previously analyzed in the NAC-1 SAR (NAC 1990). The presence of the ISO shipping container was not included in the analysis. This assumption is conservative since the shipping container is assumed to act as protection for the cask by absorbing some of the energy and distributing the load imparted to the cask during the crane boom drop. The analysis of the crane boom drop concludes that the crane boom will fail before it can impart sufficient load on the cask to breach confinement. Local lacing members in the boom fail at the point of impact and the boom structure collapses, relieving the load and subsequent energy imparted to the cask.
Cask Energy Absorption. The response of the cask to the 30-ft side impact drop is appropriate for comparison to the boom drop effects. In a side drop, the SAR analyses show that the cask top and bottom end ring support structures, along with the neutron shield chamber, can absorb approximately $23 \times 10^6$ in-lbs of energy without breach of confinement. These members are considered totally sacrificial structures and energy absorption is achieved by means of considerable crushing of these structures. The crane boom is 80 ft long and weighs 22,490 lbs. The potential energy of the boom falling from the vertical position is $11 \times 10^6$ in-lbs. The reaction forces on the cask at the end ring support structures for the crane boom drop are therefore enveloped by the existing 30-ft drop accident analysis.

Cask Bending Moment. The maximum bending moment of the cask was also evaluated during the crane boom drop event and compared with the bending moment imparted to the cask during the 40-in. puncture drop (SNF-4794). Impact on the midspan of the cask was considered the most critical point for beam-bending of the cask. The analysis of the cask bending moment during the crane boom drop shows that the maximum reaction force the boom can impart on the cask during impact is $2.9 \times 10^5$ lbs at the mid-point of the crane boom. This value compares with an allowable force of $6.8 \times 10^5$ lbs that would be required to impart the equivalent bending moment analyzed in the 40-in. drop analysis. Thus, the bending moment of the cask during the crane boom drop event is approximately a factor of two times less than that found acceptable in the previous NAC-1 SAR analysis.

Cask Puncture Resistance - Crane Boom. The cask’s resistance to puncture during the crane boom drop event was also analyzed (SNF-4794). The analysis concluded that the maximum reaction force that the crane boom could impart to the cask was $2.9 \times 10^5$ lbs. This compares to the calculated value of $6.9 \times 10^5$ lbs required for the bottom cord members of the crane boom to puncture the cask’s ¾-in. outer shell. Thus, puncture of the cask’s outer shell by the crane boom is not expected.

Cask Puncture Resistance - Load Block. An analysis was performed to evaluate the ability of the 4,000-lb load block at the end of the crane boom to puncture the cask (SNF-4794). The load block has a massive hook with about a 2.5-in. leading edge that results in an initial contact area between the block and the cask that is less than the area of the 6-in. diameter pin in the 40-in. drop accident. The maximum lift height of the load block above the cask is 75 ft. The results of the analysis conclude that although the cask will suffer physical damage, the integrity of the cask will be maintained in the maximum block height condition. Conservatively, a value of 56 ft will be used as limiting criteria for the block height over a cask/ISO container system (SNF-4794). This height provides sufficient overhead clearance to perform all rigging and hoisting activities at the ISA.

Cask Puncture Resistance - Hydraulic Spreader Bar. An analysis was performed to evaluate the capability of the 20,610-lb hydraulic spreader bar, connected at the crane hook, to puncture the cask if accidentally dropped (SNF-4794). The hydraulic spreader bar is a massive lifting fixture for the ISO containers. It has a frame of large tubular channels that allow it to
telescope in length. It also has attached positioning guides at each corner that protrude downward 15 in. beneath the corner connecting devices. These alignment plates are 0.75 in. thick by 8 in. across. The analysis evaluated bending loads from a drop accident where the fixture lands at a single point on its side, and then assessed the puncture potential from a flat drop that lands on a single alignment plate as the worst-case leading edge for the puncture. The results of the analysis conclude that bending is the controlling parameter that establishes the requirement for a maximum permissible lift height. Although the cask may suffer physical damage, the confinement integrity of the cask will be maintained by imposing a maximum height condition of 6 ft above the ISO container lid, approximately 14.75 ft above the ground (SNF-4794). This height provides sufficient overhead clearance to perform all rigging and hoisting activities at the ISA.

The analysis of crane boom, load block drop, and hydraulic spreader bar drop scenarios do not take credit for the presence of the ISO shipping container. The cask, as mounted in the ISO shipping container, is designed to shear at the lifting and rotation trunnions during an impact event, while maintaining cask integrity (NAC 1990). Since the ISO shipping container is not required to survive the crane boom and load block drop accidents, it provides no safety function for this accident.

D3.4.2.2.2 Source Term Analysis. Current analyses for the FFTF and LWR NAC-1 casks demonstrate that complete confinement of the SNF will be maintained during and after the analyzed crane boom fall. No source term analysis is required for the ISC or NAC-1 cask.

The source term analysis developed for the unmitigated TRIGA accident in Section D3.4.1.1 is applicable for this scenario.

The events and analysis depend upon initial conditions; however, these conditions can be violated during operations. For example, a crane might be taken out of service and replaced with a different crane, which would represent an unanalyzed condition. If a boom drop were to occur with a different boom, catastrophic damage is not expected because of the margin and conservatisms in the analyses. But since the package responses are unknown, a boom drop from a crane that has not been analyzed in the scenario development is considered bounded by the source terms calculated for the worst-case, non-mechanistic release shown in Section D3.4.1 for each fuel type.

D3.4.2.2.3 Consequence Analysis. The postulated crane fall accident consequences have been prevented by the passive design features of the ISC and the NAC-1 cask. A consequence analysis is not needed for the FFTF or LWR NAC-1 casks.

The consequence developed for the unmitigated TRIGA accident in Table D3-10 is applicable for this scenario.
D3.4.2.2.4 Comparison to Guidelines.

Unmitigated event – For events that exceed the initial conditions of the analyses, the bounding consequences for each fuel type show that the off-site release limits are not challenged. Therefore, no safety-class confinement SSCs are necessary.

For the NRF TRIGA fuel, the on-site consequences do not exceed the on-site risk evaluation guidelines. Therefore, for the purposes of controlling accident releases following a handling or drop accident, no safety-significant SSCs are required to confine the NRF TRIGA fuel. However, the NRF TRIGA casks do provide structural protection of the SNF payload in the event of a drop, and as such, are designated as important-to-safety Category B for NRC equivalency. On this basis, the NRF TRIGA casks are designated as Safety Significant.

For the FFTF fuel and the LWR fuel in the NAC-I casks, the bounding on-site consequences for drops outside of the initial conditions analyzed exceed the on-site risk evaluation guidelines. Therefore, safety-significant SSCs are required to prevent or mitigate the dose consequences.

Mitigated event – The postulated crane fall accident consequences have been prevented by the passive design features of the FFTF and NAC-I casks; therefore, there is no release from this event.

For TRIGA SNF, the consequence analysis was performed for the 200 Area ISA based on the unmitigated release discussed in Section D3.4.1.2. Using appropriate air transport and unit dose factors, this particulate has been used to calculate on-site and off-site receptor dose estimates. The total off-site and on-site individual doses are estimated to be 3.4E-06 rem and 9.01E-03 rem, respectively.

The radiological dose consequences from the crane fall accident for the 200 Area ISA are below the off-site release limits and the on-site dose consequence guidelines; therefore, no safety-class or safety-significant equipment is required to mitigate the potential effects of this accident for TRIGA fuel.

D3.4.2.2.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls. The following SSCs and TSR controls ensure that the SNF is adequately confined and protected from mobile crane fall accidents.
Fast Flux Test Facility Fuel

The following SSC is designated as Safety Significant:

- **FFTF ISC** – The ISC is designated as Safety Significant to maintain confinement of radioactive materials after the crane fall and to provide passive protection of the CCC such that it retains structural integrity.

The following TSR control is necessary to protect the assumptions made in the mobile crane fall analyses:

- **Lift restriction** – A TSR shall be in place to restrict crane use to that bounded by the calculational analysis for the FFTF ISC. Only the Manitowoc 4000 150T crane with Model 22 80-ft boom or the Manitowoc 4100 250T crane with Model 27 80-ft boom shall be used to lift the ISC (SNF-4794).

NRF TRIGA

No SSCs credited with a preventive or mitigative function.

Light Water Reactor Fuel

The following SSC is designated as Safety Significant:

- **NAC-1 cask system** – The NAC-1 cask has been designated Safety Significant to provide passive structural integrity protection of the inner canister.

The following TSR control is necessary to protect the assumptions made in the mobile crane accident:

- **Lift restriction** – A TSR shall be in place to restrict use of the crane, lifting equipment, and lift height to that bounded by the calculational analysis for the NAC-1 casks. Only the Manitowoc 4000 150T crane with Model 22 80-ft boom or the Manitowoc 4100 250T crane with Model 27 80-ft boom shall be used to lift the ISC (SNF-4794).

D3.4.2.3 Cask Tipover.

**D3.4.2.3.1 Scenario Development.** Cask handling, operations near the storage casks, or external forces could cause an accident that would lead to the tipover and slap-down of a cask. If the impact caused a cask to undergo accelerations and forces beyond the design strength, the cask containment could be breached. Breach of the cask containment could result in radiological releases. Casks also maintain the geometry of stored SNF to meet any criticality safety concerns,
and a cask breach or deformation could lead to the violation of a criticality control limit. A summary of the safety features required to prevent or mitigate the cask tipover accident is provided in Table D3-13.

Table D3-13. Summary of Safety Features Required to Prevent Cask Tipover Accident.

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator (1)</th>
<th>Safety function</th>
<th>Safety features (described in Chapter D4.0)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cask tipover</td>
<td>G-03</td>
<td>Maintain criticality geometry control after cask is overturned. Maintain confinement of radioactive materials and provide passive protection such that structural integrity is maintained after a cask is overturned.</td>
<td>Safety class equipment for criticality geometry control: • CCC. Safety-significant equipment for confinement: • FFTF ISC • NRF TRIGA cask • 2R container. Safety-significant equipment for structural integrity: • FFTF ISC • ISO container • Rad-Vault • DOT-6M container • NAC-1 cask. Defense-in-depth: • Qualified crane operators • Detailed procedures.</td>
</tr>
<tr>
<td></td>
<td>G-06</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>G-13</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

(1) Checklist designators are from SNF-4820, 1999, 200 Area Interim Storage Area Final Hazard Analysis, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.

2R = inner container of DOT-6M.  
CCC = core component container.  
DOT = U.S. Department of Transportation.  
FFTF = Fast Flux Test Facility.  
ISC = interim storage cask.  
ISO = International Standards Organization.  
NAC = Nuclear Assurance Corporation.  
NRF = Neutron Radiography Facility.  
TRIGA = Training, Research and Isotope Production, General Atomics.
Fast Flux Test Facility

The ISC design analysis report (General Atomics 1995) describes the cask tipover accident analyses performed for the ISC. The analysis considers an unrestrained rotational fall from a vertical to a horizontal position onto a flat unyielding surface. This analysis demonstrates that the ISC will survive a tipover accident onto any flat horizontal surface. The cask forces during this accident are bounded by those for the accident analysis for a 4-ft free drop of the cask side onto an unyielding surface (General Atomics 1995). Under these conditions, the ISC will not credibly lose its confinement and criticality control geometry and, thus, will not lead to credible radioactive releases.

NRF TRIGA

The 200 Area ISA analysis performed for the Rad-Vault containing six NRF TRIGA casks and two TRIGA FFCR DOT-6M shipping containers demonstrated that the Rad-Vault will not tip over during a 360 mph design basis tornado (DBT) or seismic event (SNF-4792).

While no analysis specifically addresses a TRIGA cask tip over, drop analyses have been performed that would bound such an accident. The TRIGA cask has been analyzed for a 109-in. drop height onto concrete, with the cask in a horizontal position. The concrete used in the analysis has a thickness of 16 in., a strength of 4000 lbs/in², and is reinforced with #7 rebar that is spaced 14 in. apart and placed 2 in. below the concrete surface. A simple unrestrained tipover accident of a TRIGA cask onto any flat surface, with compressive strength less than or equal to the described concrete, is not expected to lead to a breach of cask confinement.

While no tipover testing was performed for the DOT-6M, severe drop tests were completed that will bound such an accident. The drop tests onto an unyielding surface showed no breach of confinement for a 30-ft drop. These tests were conducted with the container in a horizontal orientation (Martin Marietta 1992). Any tipover accident onto a flat horizontal surface will be bounded by the results of this horizontal drop test, which demonstrated that containment will not be breached. Under these conditions, the DOT-6M will not credibly lose its confinement and will not lead to credible radioactive releases.

Light Water Reactor Fuel

While an unrestrained tipover accident has not been analyzed, the NAC-1 cask was analyzed previously for a 30-ft free drop onto a flat, horizontal unyielding surface for end, side, and corner impact orientations (NAC 1990, Section 2.7.1). The NAC-1 cask will not breach from this 30-ft drop for any orientation, and the analysis is expected to easily bound any consequences of a simple unrestrained tipover onto any flat surface. Under these conditions, the NAC-1 cask will not credibly allow damage to the LWR canister resulting in loss of criticality control geometry or credible radioactive releases.
D3.4.2.3.2 Source Term Analysis.

Fast Flux Test Facility

All ISA analyses demonstrate that complete confinement of the SNF will be maintained during and after the postulated and analyzed tipover events. All current analyses assumed that the impact surface is unyielding. The ISA storage pad will yield to the impact of an FFTF cask drop, and absorb drop energy at least as well as the analyzed concrete. The tipover or drop analysis assumes, in general, that the drop surface is flat. The ISA storage pad and any areas that a cask could tip over onto should be flat. The confinement of each cask will be maintained in each of the considered tipover events and, therefore, a source term analysis is not required.

NRF TRIGA

The Rad-Vault has not been analyzed for a tipover accident, but has been demonstrated to not tip during a DBT or seismic event. A tipover accident is not considered credible for the Rad-Vault in the 200 Area ISA.

Light Water Reactor Fuel

The existing ISA analyses demonstrate that complete confinement of the SNF will be maintained during and after the postulated and analyzed tipover events.

D3.4.2.3.3 Consequence Analysis.

Fast Flux Test Facility

The postulated tipover accident consequences have been prevented by the passive design features of the FFTF ISC. A consequence analysis is not needed.

NRF TRIGA

The postulated tipover accident consequences have been prevented by the passive design features of the Rad-Vault and each TRIGA container. A consequence analysis is not needed.

Light Water Reactor Fuel

The postulated tipover accident consequences have been prevented by the passive design features of each NAC-1 cask. A consequence analysis is not needed.
D3.4.2.3.4 Comparison to Guidelines.

**Unmitigated event** – For scenarios with conditions that exceed the initial conditions of the analyses, the bounding consequences for each fuel type show that the off-site release limits are not challenged. Drop analyses have been performed that would bound the cask tipover accident. Therefore, no safety-class confinement SSCs are necessary.

For the FFTF fuel and the LWR fuel in the NAC-1 casks, the bounding on-site consequences for drops outside of the initial conditions analyzed exceed the on-site risk evaluation guidelines. Therefore, safety-significant SSCs are required to prevent or mitigate the dose consequences.

For the NRF TRIGA fuel, the on-site consequences do not exceed the on-site risk evaluation guidelines. Therefore, for the purposes of controlling accident releases following a handling or drop accident, no safety-significant SSCs are required to confine the NRF TRIGA fuel. However, the NRF TRIGA casks do provide structural protection of the SNF payload in the event of a drop, and as such, are designated as important-to-safety Category B for NRC equivalency. On this basis, the NRF TRIGA casks are designated as Safety Significant.

While no safety-class SSCs are required to provide confinement, some SSCs are required to protect the geometric configuration of the fuel for criticality safety. SSCs credited with retaining the critically safe geometry of the fuel are designated as Safety Class.

**Mitigated event** – For the tipover events postulated and analyzed in the scenario development, the potential releases are prevented. These events take credit for the packaging components and limitations on handling (e.g., lift height) that have been shown to provide the impact protection and confinement of the fuel for the tipover drop events.

D3.4.2.3.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls. The following SSCs and TSR controls ensure that the SNF is adequately confined and protected from inadvertent nuclear criticality during and following a tipover event.

**Fast Flux Test Facility Fuel**

The following SSCs are designated as Safety Significant or Safety Class:

- **FFTF ISC** – The ISC is designated as Safety Significant to maintain confinement of radioactive materials after the tipover and to provide passive protection of the CCC such that it retains structural integrity.

- **CCC** – The CCC is designated as Safety Class to maintain structural integrity to provide criticality geometry control.
NRF TRIGA

The Rad-Vault has been analyzed to not tip over or slide as a result of NPH or DBAs.

Light Water Reactor Fuel

The NAC-1 cask in the ISO container has been analyzed to not tip over or slide as a result of NPH or DBAs.

D3.4.2.4 Fuel Rod Rupture.

D3.4.2.4.1 Scenario Development. Fission gases generated in the SNF are contained by the fuel cladding. Deterioration of the clad by oxidation or stress cracking could result in a release of the contained fission gas to the storage container. If the design pressure value for the sealed storage container is exceeded, these fission gases could be released to the environment. A summary of the safety features required to prevent the fuel-rod rupture accident is provided in Table D3-14.

Fast Flux Test Facility Fuel

Based on the ISC design analysis report (General Atomics 1995, page 4.1-4), the cavity internal pressure for normal long-term conditions is 33.9 lbs/in² gauge, based on an average fuel cavity temperature of 245 °F. The maximum pressure occurs for volcanic ash accumulation with plugged ducts and is 36.7 lbs/in² gauge based on the highest fuel cavity temperature of 286 °F (WHC-SD-FF-ER-100). These pressures are calculated at the end of the cask’s 50-year life assuming fission gas generation and no reduction in decay heat. Per Specification WHC-S-4110, Specification for FFTF Interim Storage Cask, Section 3.2.2.3, the generation of gases produced by radioactive decay is expected to be 5.4 gram-atoms (moles) per assembly or 37.8 gram-atoms per ISC for a 50-year period, assuming that all free gases contained within the fuel rod cladding are released. The design pressure is based on a CCC containing seven fuel assemblies.
Table D3-14. Summary of Safety Features Required to Prevent Fuel Rod Rupture Accident.

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator (1)</th>
<th>Safety function</th>
<th>Safety features (described in Chapter D4.0)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Uncontrolled release of gases from fuel rod rupture from casks</td>
<td>H-06 H-11</td>
<td>Maintain structural integrity for criticality geometry control after pressurization. Maintain confinement after rupture of all fuel pins.</td>
<td>Safety class equipment for criticality geometry control: • CCC • LWR canister. Safety-significant equipment for confinement: • FFTF ISC • NRF TRIGA cask • 2R container • LWR canister. TSRs: • Restriction on minimum spacing between ISCs. • Restriction on maximum fuel loading in the FFTF ISC and LWR canister.</td>
</tr>
</tbody>
</table>

(1) Checklist designators are from SNF-4820, 1999, 200 Area Interim Storage Area Final Hazard Analysis, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.

2R = inner container of DOT-6M.
CCC = core component container.
FFTF = Fast Flux Test Facility.
ISC = interim storage cask.
LWR = light water reactor.
NRF = Neutron Radiography Facility.
TRIGA = Training, Research and Isotope Production, General Atomics.
TSR = Technical Safety Requirement.
These results have assumed that the casks are spaced no closer than 5.5 ft from surface to surface. Smaller spacings will cause temperatures to rise. If, for example, the casks are spaced 24 in. in one direction and 44 in. in the other, the temperature will increase by approximately 5 to 10 °F. This is due partly to the restricted radiant heat transfer to the sky and partly to the increase in ambient temperature at the interior of the array. All temperatures will still be within limits, but margins will decrease accordingly. The planned spacing for the 200 Area ISA is 4 ft in both directions. The temperature variation from 5.5-ft to 4-ft spacing was not reevaluated, as the tightened array described above had minimal effect. No temperature limits will be exceeded if the cask is positioned on its side due to an accidental drop or tipover. Although the ventilation flow will be ineffective, the situation will be no worse than the case of ash accumulation with plugged ducts, with temperatures remaining within the limits.

**NRF TRIGA**

The NRF TRIGA cask design pressure is 11.2 lbs/in² gauge. The design pressure is based on the maximum pressure differential obtained from the 10 CFR 71.71(c)(3) reduced external pressure of 3.5 lbs/in² absolute. The DOT-6M/2R design pressure is 40 lbs/in² gauge. The NRF TRIGA cask was pressure-tested at 17.3 lbs/in² gauge, and the DOT-6M/2R container was pressure-tested at 60 lbs/in² gauge. The pressure boundary for the DOT-6M/2R is the 2R container.

The low burnup of the SNF has produced a small quantity of fission products. Most of the fission product gases are captured within the fuel matrix. The fission product gases within each element (25 lbs/in² maximum initial pressure) will result in negligible pressure in the cask or container (less than 2 lbs/in²), if all of the fuel elements are breached within the cask (WHC-SD-FF-TI-043). By-products from elastomeric seal degeneration will have minimal effect. Therefore, it is expected that the NRF TRIGA cask pressure will remain less than the design limit of 11.2 lbs/in² gauge.

**Light Water Reactor**

Analysis determined the maximum canister pressures, due to fuel-rod rupture of worst-case hypothetical pressurized water reactor (PWR) and boiling water reactor (BWR) SNF assemblies for both normal and fire accident conditions. Analyses using the existing design of the NAC-1 cask (NAC 1990) determined the maximum cask pressures due to fuel-rod rupture of a worst-case hypothetical PWR SNF assembly for both normal and fire accident conditions (SNF-4794). The pressure within the inner canister cavity due to rod rupture at maximum normal conditions for consolidated BWR and PWR fuel rods is 30.86 lbs/in² absolute, which is 16.16 lb/in² gauge. This is within the design pressure of 50 lb/in² gauge and test pressure of 85 lb/in² gauge established for the inner canister. This also bounds the consolidated BWR and PWR rod normal conditions with the NAC-1 cask cavity that has a design pressure of 65 lb/in², which is tested to 100 lb/in².
Worst case consolidated fuel rod canister pressures were also calculated for 100% rod rupture during the design basis fire conditions (SNF-4794) resulting in an inner canister cavity maximum average temperature of 540 °F. The resultant pressure at this temperature for the consolidated BWR and PWR fuel rods is 22.44 lb/in². This consolidated BWR and PWR rod accident condition pressure is also well within the design pressures for both the inner canister and the NAC-1 cask.

D3.4.2.4.2 Source Term Analysis. No source term is calculated because no release of radioactive material is identified for the conditions discussed in the subsections that follow.

Fast Flux Test Facility

The internal storage container pressure calculated for the release of all fission gases from the fuel rods is within the design pressure of the storage vessel.

NRF TRIGA

For TRIGA SNF, the internal storage container pressure calculated for the release of all fission gases from the fuel rods is within the design pressure of the storage vessel. Consequently, the casks will not release radioactive material.

Light Water Reactor Fuel

For the LWR fuel, the internal storage container pressure calculated for the release of all fission gases from the fuel rods is within the design pressure of the storage vessel.

D3.4.2.4.3 Consequence Analysis. The postulated fuel rod rupture consequences have been prevented by the passive design features of the SNF casks. A consequence analysis is not needed.

D3.4.2.4.4 Comparison to Guidelines.

Unmitigated event – For events that exceed the initial conditions of the analyses, the bounding consequences for each fuel type show that the off-site release limits are not challenged. Therefore, no safety-class confinement SSCs are necessary.

For the FFTF fuel and the LWR fuel, the bounding on-site consequences for hypothetical releases outside of the initial conditions analyzed exceed the on-site risk evaluation guidelines. Therefore, safety-significant SSCs are required to prevent or mitigate the dose consequences.

For the NRF TRIGA fuel, the on-site consequences do not exceed the on-site risk evaluation guidelines. Therefore, for the purposes of controlling accident releases following fuel rod rupture, no safety-significant SSCs are required to confine the NRF TRIGA fuel. However,
the NRF TRIGA casks do provide structural protection of the SNF payload, and as such, are designated as important-to-safety Category B for NRC equivalency. On this basis, the NRF TRIGA casks are designated as Safety Significant.

While no safety-class SSCs are required to provide confinement, some SSCs are required to protect the geometric configuration of the fuel for criticality safety. SSCs credited with retaining the critically safe geometry of the fuel are designated as Safety Class.

**Mitigated event** – The postulated accidents have been prevented by the passive design features of the casks; therefore, there is no release from this event.

**D3.4.2.4.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls.** The following SSCs and TSR controls ensure that the SNF is adequately confined and protected from heat loads, and keep the cask internal pressure within design values.

**Fast Flux Test Facility Fuel**

The following SSCs are designated as Safety Significant or Safety Class:

- **FFT F ISC** – The ISC is designated as Safety Significant to maintain confinement of the radioactive materials after the rupture of all fuel pins.

- **CCC inner container** – The CCC is designated as Safety Class to maintain structural integrity to provide criticality geometry control at the pressure generated from the rupture of all fuel pins.

The following TSR controls are necessary to protect the assumptions made in the cask internal pressure analyses:

- A minimum spacing array area shall be no less than 24 in. by 44 in.
- There shall be no more than seven fuel assemblies in a CCC.

**NRF TRIGA**

The following SSCs are designated as Safety Significant:

- **NRF TRIGA casks** – The TRIGA casks are designated as Safety Significant to maintain confinement of the radioactive materials after the rupture of all fuel pins.

- **2R container** – The 2R containers are designated as Safety Significant to maintain confinement of the radioactive materials after the rupture of all fuel pins.
Light Water Reactor Fuel

The following SSC is designated as Safety Significant and Safety Class:

- **LWR canister** – The LWR canister is designated as Safety Significant to maintain confinement of the radioactive materials and Safety Class to maintain structural integrity to provide criticality geometry control at the pressure generated.

The following TSR control is necessary to protect the assumptions made in the cask pressure analysis:

- The number of individual fuel rods in the LWR canister does not exceed a maximum of 179 PWR rods or 96.5 BWR rods consolidated with 17 PWR rods.

**D3.4.2.5 Seismic.**

**D3.4.2.5.1 Scenario Development.** A summary of the safety features required to prevent consequences after the seismic accident is provided in Table D3-15.

While the SNF storage systems are located at the 200 East Area, a design basis earthquake (DBE) could occur. The SNF storage systems could sustain seismically induced accelerations and forces, or impacts with nearby obstacles by overturning or sliding, which could potentially damage the structure of the storage system. These seismic accelerations and forces could affect the structural integrity or functionality of the SNF storage systems and potentially release radioactive materials out of the SNF storage system. All seismic g forces are well bounded by the drop accelerations.

**Fast Flux Test Facility**

200 Area ISA site-specific analyses (HNF-2183) determined the margins of safety of the ISC in the 200 East Area against overturning and sliding during a seismic event. Other analyses were previously performed for a lower level earthquake than that currently considered for the 200 East Area (General Atomics 1995). Under the 200 East Area site-specific loadings (zero-period acceleration [ZPA] horizontal spectra of 0.26 g and vertical of 0.18 g, with a second earthquake of horizontal ZPA of 0.50 g and a vertical ZPA of two-thirds of the horizontal), the normal storage condition of the ISC was found to have a margin of safety against overturning of +0.20 (a factor of safety of +1.20) and against sliding of +0.33 (a factor of safety of +1.33). NRC criteria state that a minimum factor of safety of +1.1 must be maintained against sliding and overturning during a seismic event. These calculations found that the ISC had sufficient margins of stability during the DBE to not overturn and not slide.
Table D3-15. Summary of Safety Features Required to Withstand Seismic Accident.

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator</th>
<th>Safety function</th>
<th>Safety features (described in Chapter D4.0)</th>
</tr>
</thead>
</table>
| Natural phenomena - seismic event | R-01 | Withstand seismic accelerations without loss of confinement, tipover, or sliding. | Safety class equipment for criticality geometry control:  
  • CCC  
  • LWR canister.  
Safety-significant equipment for confinement:  
  • FFTF ISC  
  • NRF TRIGA cask  
  • 2R container  
  • LWR canister.  
Safety-significant equipment for seismic stability/structural integrity:  
  • FFTF ISC  
  • ISO container  
  • Rad-Vault  
  • DOT-6M container  
  • NAC-1 cask  
  • ISO container.  
TSR:  
  • FFTF ISC and NAC-1 cask placed on concrete pad.  
  • Rad-Vault placed on compacted gravel.  
Defense-in-depth:  
  • Canister Storage Building personnel are trained in sitewide and facility-specific emergency response. |

(1) Checklist designators are from SNF-4820, 1999, 200 Area Interim Storage Area Final Hazard Analysis, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.

**Abbreviations:**
- 2R = inner container of DOT-6M.  
- CCC = core component container.  
- DOT = U.S. Department of Transportation.  
- FFTF = Fast Flux Test Facility.  
- ISC = interim storage cask.  
- ISO = International Standards Organization.  
- LWR = light water reactor.  
- NAC = Nuclear Assurance Corporation.  
- NRF = Neutron Radiography Facility.  
- SSC = structure, system, and component.  
- TRIGA = Training, Research and Isotope Production, General Atomics.  
- TSR = Technical Safety Requirement.
Previous 400 Area ISA analyses (General Atomics 1995) also evaluated for overturning by simultaneously applying static loads to the ISC in the horizontal and vertical directions. The seismic motions from the DBE (0.25 g) earthquake were evaluated. These design response spectra were applied for the horizontal motion in the free field. For the vertical motion, the design response spectra were taken as two-thirds the response spectra for the horizontal motion. This analysis found that the ISC has a margin of safety of 0.29 against overturning during the DBE seismic event. Additionally, even if the seismic g level was higher and the ISC were to tip over, the tipover accident analysis performed for the ISC found that the cask still retains the stored SNF in a safe configuration.

Movement of the ISC during an earthquake was evaluated. The analysis of the ISC for sliding shows that a coefficient of friction of at least 0.3 is required to prevent sliding motion. This result was compared to the coefficient of friction of concrete on concrete as determined from the ACI Manual of Concrete Practice (ACI 1989, Part 4, Section 11.7.4.3). Since this value is 0.6, there is a large margin against sliding motion during an earthquake.

**TRIGA**

Using the current 200 East Area site-specific loadings, analyses were performed that demonstrate structural margins exist for the 200 Area ISA movement. The analyses (SNF-4792) found that the design of the Rad-Vault is well within the DBE for the site, and seismic stability is assured. The Rad-Vault will not overturn or slide on the soil. The lid lip will not fail, and the lid will remain on the Rad-Vault. The NRF TRIGA casks inside the container will not suffer loss of integrity, as their design for use in ground transport provides capacities far in excess of 0.26 g (WHC-SD-TP-SARP-008). Likewise, the DOT-6M containers that will be used for the FFCR are qualified for conditions that are in excess of the 0.26 g ground acceleration earthquake analyzed for the 200 Area ISA (SNF-4792).

The margin of safety for overturning the loaded Rad-Vault is 1.1. Sliding was also examined using a friction factor of 0.35 between the gravel and the concrete base of the Rad-Vault. This factor was taken from the values shown in the Uniform Building Code for sandy gravel and/or gravel (ICBO 1994). The margin of safety against sliding was 0.104.

The soil-bearing pressure beneath the Rad-Vault was not exceeded under seismic overturning loads. The margin of safety for soil bearing pressure was 0.59. As such, the Rad-Vault is not expected to overturn due to soil compaction loads.

**Light Water Reactor**

Current seismic criteria is specified in Table D2-2. Calculations were performed for the 200 Area ISA to evaluate the cask/container’s resistance to overturning during the DBE (SNF-4794). The margin of safety against overturning the cask/container unit onto its left or right side is 2.7. The margin of safety against overturning the cask/container unit onto its front...
or back is 3.9. Based on these results, it was concluded that the cask/container units will not overturn.

Sliding of the cask/container was also examined using a friction factor of 0.5 between the concrete and the bottom side of the ISO shipping container. The margin of safety against sliding was found to be 0.58, which indicates that the cask/container will not slide.

Tipping of the storage system due to soil compaction beneath the pad under seismic overturning loads was also calculated for the 400 Area ISA. The margin of safety is 4.7.

**D3.4.2.5.2 Source Term Analysis.** No source term is calculated because no release of radioactive material is identified.

**D3.4.2.5.3 Consequence Analysis.** The consequences from the postulated seismic event are prevented by the passive design features of the SNF cask. A consequence analysis is not needed for the FFTF, TRIGA, or LWR fuel storage systems.

**D3.4.2.5.4 Comparison to Guidelines.**

**Unmitigated event** – For drops that exceed the initial conditions of the analyses, the bounding consequences for each fuel type show that the off-site release limits are not challenged. Drop analyses have been performed that would bound the seismic event. Therefore, no safety-class confinement SSCs are necessary.

For the NRF TRIGA fuel, the on-site consequences do not exceed the on-site risk evaluation guidelines. Therefore, for the purposes of controlling accident releases following a seismic event, no safety-significant SSCs are required to confine the NRF TRIGA fuel. However, the NRF TRIGA casks do provide structural protection of the SNF payload, and as such, are designated as important-to-safety Category B for NRC equivalency. On this basis, the NRF TRIGA casks are designated as Safety Significant.

For the FFTF fuel and the LWR fuel in the NAC-I casks, the bounding on-site consequences for a seismic event outside of the initial conditions analyzed exceed the on-site risk evaluation guidelines. Therefore, safety-significant SSCs are required to prevent or mitigate the dose consequences.

**Mitigated event** – For the seismic events postulated and analyzed in the scenario development, the potential releases are prevented. These events take credit for the packaging components and limitations on handling (e.g., lift height) that have been shown to provide the impact protection and confinement of the fuel for the analyzed seismic events.
D3.4.2.5.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls. The following SSCs and TSR controls ensure that the SNF is adequately confined and protected from the seismic event within design values.

Fast Flux Test Facility Fuel

The following SSCs are designated as Safety Significant or Safety Class:

- **FFTF ISC** – The ISC is designated as Safety Significant to withstand seismic accelerations without loss of confinement or structural integrity, tipover or sliding, or to survive a tipover accident.

- **CCC** – The CCC is designated as Safety Class to withstand seismic accelerations without loss of structural integrity for geometry control.

The following TSR control is necessary to protect the assumptions made in the seismic analyses:

- The FFTF ISC shall be placed on a concrete pad.

NRF TRIGA

The following SSCs are designated as Safety Significant or Safety Class:

- **NRF TRIGA casks** – The TRIGA casks are designated as Safety Significant for NRC important-to-safety considerations for providing confinement for the SNF payload.

- **Rad-Vault** – The Rad-Vault is designated at Safety Significant for seismic stability and structural integrity protection.

- **DOT-6M containers** – The DOT-6M containers are designated as Safety Significant for NRC important-to-safety considerations for providing seismic stability and structural protection of the SNF payload.

- **2R container** – The 2R containers are designated as Safety Significant for NRC important-to-safety considerations to maintain confinement of radioactive materials.

The following TSR control is necessary to protect the assumptions made in the seismic event:

- The Rad-Vault shall be placed on compacted gravel.
Light Water Reactor Fuel

The following SSCs are designated as Safety Significant or Safety Class:

- **LWR canister** – The LWR canister is designated as Safety Class to provide a critically safe geometry for the SNF payload. In addition, the LWR canister provides a Safety Significant confinement function for the SNF payload.

- **ISO container** – The ISO container is designated as Safety Significant for seismic stability and structural integrity protection.

The following TSR control is necessary to protect the assumptions made in the seismic event:

- The NAC-1 cask shall be placed on a concrete pad.

**D3.4.2.6 Tornado/Wind.**

**D3.4.2.6.1 Scenario Development.** While the SNF storage systems are located at the 200 East Area, DBAs of straight wind and tornado wind could occur. Tornado missiles are not analyzed since probabilistic risk assessment analyses have concluded that tornado missiles do not need to be considered (HNF-1785). The SNF storage systems could sustain wind loadings or wind effects that could move the SNF storage systems to impact with nearby obstacles. Additionally, the SNF storage systems could suffer impacts from wind-driven missiles. These effects could affect the structural integrity or functionality of the SNF storage systems and potentially release radioactive materials out of the SNF storage system. A summary of the safety features required to prevent tornado and wind accident releases is provided in Table D3-16.

**Fast Flux Test Facility**

Analyses using the current 200 East Area site-specific requirements have been performed (SNF-4790). Calculations show that the loaded ISC will not tip over or slide. The increase in internal pressure due to differential pressure loading is negligible. Wind-driven missiles result in concrete penetration of less than 1 in.

**TRIGA**

Supplemental analyses using the current 200 East Area site-specific requirements have been performed demonstrating adequacy of the Rad-Vault to withstand the DBT and the wind-driven missile (SNF-4792).
Table D3-16. Summary of Safety Features Required to Prevent Tornado/Wind Accident.

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator (1)</th>
<th>Safety function</th>
<th>Safety features (described in Chapter D4.0)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Natural phenomena - tornado and wind</td>
<td>R-06, R-08</td>
<td>Withstand DBT winds and the design basis wind and wind-driven missiles without loss of confinement.</td>
<td>Safety-significant equipment for confinement:</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Without sliding or tip over.</td>
<td>• FFTF ISC</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• NRF TRIGA cask</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• 2R container</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• LWR canister.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Safety-significant equipment for structural integrity:</td>
<td>• FFTF ISC</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Rad-Vault</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• NAC-I cask</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• ISO container.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>TSR:</td>
<td>• FFTF ISC and NAC-I cask placed on concrete pad.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Rad-Vault placed on compacted gravel.</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Defense-in-depth:</td>
<td>• Operations personnel are trained in sitewide and facility-specific emergency response.</td>
</tr>
</tbody>
</table>

(1) Checklist designators are from SNF-4820, 1999, 200 Area Interim Storage Area Final Hazard Analysis, Rev. 0, Fluor Daniel Hanford, Incorporated, Richland, Washington.

Legend:
- 2R = inner container of DOT-6M
- DBT = design basis tornado
- FFTF = Fast Flux Test Facility
- ISC = interim storage cask
- ISO = International Standards Organization
- LWR = light water reactor
- NAC = Nuclear Assurance Corporation
- NRF = Neutron Radiography Facility
- TRIGA = Training, Research and Isotope Production, General Atomics
- TSR = Technical Safety Requirement
Light Water Reactor

Analyses were performed for the 200 Area for a tornado wind speed of 200 mph and a differential pressure of 0.90 lbs/in² over 3 seconds. The NAC-1 cask inside the ISO container does not slide or tip (SNF-4794). Wind missile damage does not reduce confinement.

D3.4.2.6.2 Source Term Analysis. For spent fuel storage, the calculations determined that the design criteria were met and no releases are expected to occur as a result of this event. As such, source term analysis is not required.

D3.4.2.6.3 Consequence Analysis. The consequences from the postulated tornado wind and wind-driven missile events are prevented by the passive design features of the robust SNF cask systems. A consequence analysis is not needed for the FFTF, TRIGA, or LWR fuel storage systems.

D3.4.2.6.4 Comparison to Guidelines.

Unmitigated event – For the bounding tornado/wind events, consequences for each fuel type show that the off-site release limits are not challenged. Therefore, no safety-class confinement SSCs are necessary.

For the NRF TRIGA fuel, the on-site consequences do not exceed the on-site risk evaluation guidelines. Therefore, for the purposes of controlling accident releases following bounding tornado/wind events, no safety-significant SSCs are required to confine the NRF TRIGA fuel. However, the NRF TRIGA casks do provide structural protection of the SNF payload, and as such, are designated as important-to-safety Category B for NRC equivalency. On this basis, the NRF TRIGA casks are designated as Safety Significant.

For the FFTF fuel and the LWR fuel in NAC-1 casks, the bounding on-site consequences for the hypothetical accident outside of the initial conditions analyzed exceed the on-site risk evaluation guidelines. Therefore, safety-significant SSCs are required to prevent or mitigate the dose consequences.

Mitigated event – For the tornado/wind events postulated and analyzed in the scenario development, the potential releases are prevented. These events take credit for the packaging components that have been shown to provide the impact protection and confinement of the fuel for the analyzed bounding tornado/wind events.

D3.4.2.6.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls. The following SSCs and TSR controls ensure that the SNF is adequately confined and protected from the bounding tornado/wind events within design values.
Fast Flux Test Facility Fuel

The following SSC is designated as Safety Significant:

- **FFTF ISC** – The ISC is designated as Safety Significant to withstand DBT winds (excluding DBT missiles) without sliding or tip over. The ISC must also withstand the design basis wind and wind-driven missiles established for the 200 Area ISA without loss of confinement, as well as provide structural integrity protection.

The following TSR control is necessary to protect the assumptions made in the tornado/wind event analyses:

- The FFTF ISC shall be placed on a concrete pad.

**NRF TRIGA**

The following SSCs are designated as Safety Significant:

- **NRF TRIGA cask** – The NRF TRIGA cask is designated as Safety Significant for confinement.

- **Rad-Vault** – The Rad-Vault is designated as Safety Significant for NRC important-to-safety considerations to withstand DBT winds (excluding DBT missiles) without sliding or tip over. The Rad-Vault also withstands the design basis wind and wind-driven missiles established for the 200 Area ISA.

- **2R container** – The 2R container is designated as Safety Significant for confinement.

The following TSR control is necessary to protect the assumptions made in the tornado/wind:

- The Rad-Vault shall be placed on compacted gravel.

**Light Water Reactor Fuel**

The following SSCs are designated as Safety Significant:

- **ISO container** – The ISO container is designated as Safety Significant for structural integrity.

- **NAC-1 cask system** – The NAC-1 cask is designated as Safety Significant to withstand DBT winds (excluding DBT missiles) without sliding or tip over. The NAC-1 cask system must also withstand the design basis wind and wind-driven
missiles established for the 200 Area ISA without structural damage to the inner canister.

- **LWR canister** - The LWR canister is designated as Safety Significant for confinement.

The following TSR control is necessary to protect the assumptions made in the tornado/wind events:

- The NAC-1 cask shall be placed on a concrete pad.

**D3.4.2.7 Fire.**

**D3.4.2.7.1 Scenario Development.** A fire of unspecified origin can be assumed to occur, which subjects the exterior of the SNF storage containers to a high, short-term heat load that could ultimately threaten the confinement capability of the containers. A summary of the safety features required to prevent the design basis fire accident is provided in Table D3-17. The fire hazard analysis (SNF-4932) concluded that all fire scenarios are bounded by the transportation fire event. The transportation fire event is defined as an exposure of the exterior surface of the container system to a temperature of 1,475 °F for a period of 30 minutes.

**Fast Flux Test Facility**

In previous 400 Area ISA analyses (General Atomics 1995), the cask was evaluated for a hypothetical accident condition resulting from an external fire. In this transient case, the cask was subjected to a radiative environment of 1,475 °F with an emissivity of 0.9. After 30 minutes of exposure, the environment returned to normal ambient temperature, with no artificial cooling applied. The initial conditions for the transient were the steady-state temperatures for long-term normal conditions. In this accident case, the hot gas is assumed to enter the ventilation ducts and flow up the annulus between the liner and inner shield. The same flow model was used as for long-term normal conditions, except that the inlet temperature during the 30 minutes was taken to be 1475 °F instead of 94 °F. The fuel cavity average temperature calculated for this case was 261 °F, resulting in a cavity pressure that is below the design pressure seal temperature limits, and the cladding temperature limits are not exceeded. Therefore, a radioactive release will not occur.
**Table D3-17. Summary of Safety Features Required to Prevent Design Basis Fire Accident.**

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Checklist designator (1)</th>
<th>Safety function</th>
<th>Safety features (described in Chapter D4.0)</th>
</tr>
</thead>
</table>
| Fire inside the ISA fence | L-03 L-07 | Withstand fire conditions without loss of confinement or exceeding temperature limits for fuel cladding or cask components. | Safety class equipment for criticality geometry control:  
  - CCC  
  - LWR canister.  
Safety-significant equipment for confinement:  
  - FFTF ISC  
  - NRF TRIGA cask (inside Rad-Vault)  
  - 2R container (inside DOT-6M container and Rad-Vault)  
  - LWR canister.  
Safety-significant equipment for structural integrity:  
  - NAC-I cask  
  - FFTF ISC  
  - Rad-Vault  
  - DOT-6M container  
  - 2R container  
  - LWR canister  
  - NRF TRIGA cask.  
TSR:  
  - Fire loadings are controlled per FHA.  
Defense-in-depth:  
  - Canister Storage Building personnel are trained in sitewide and facility-specific emergency response. |

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2R = inner container of DOT-6M.  
CCC = core component container.  
DOT = U.S. Department of Transportation.  
FFTF = Fast Flux Test Facility.  
FHA = fire hazards analysis.  
ISC = interim storage cask.  
ISO = International Standards Organization.  
NAC = Nuclear Assurance Corporation.  
NRF = Neutron Radiography Facility.  
LWR = light water reactor.  
TRIGA = Training, Research and Isotope Production, General Atomics.  
TSR = Technical Safety Requirement.
NRF TRIGA

Previous 400 Area ISA analyses (WHC-SD-FF-TI-043) assumed that during a very hot summer, a fire (e.g., range fire, spilled fuel) engulfs the concrete Rad-Vault. The conditions of the fire are such that the outer wall of the Rad-Vault experiences a temperature of 1,475 °F for a period of 30 minutes. After 30 minutes of exposure, the environment returns to normal ambient temperature, without means of artificial cooling, and the Rad-Vault commences to cool down. An analysis for the 400 Area ISA was performed to determine the peak transient temperature profile of the vault wall, the peak temperature of the inner vault wall, and the fuel element cladding maximum temperature. This analysis showed a quasi steady-state temperature for the vault wall of 177 °F. Assuming equilibrium conditions, the inner vault wall temperature will result in a fuel cladding temperature of 193 °F, where the maximum allowable fuel cladding temperature is 302 °F. Therefore, radiological material releases will not result from exceeding the temperature limits; however, the design specifications for the concrete Rad-Vault may be exceeded for a short period of time at the exterior surface and should be evaluated if this event were to occur.

Light Water Reactor Fuel

The thermal analysis for the 400 Area ISA of the NAC-1 cask subjected to the transportation fire scenario of 10 CFR 71.73(c)(3) was previously performed in the original transportation SAR (NFS 1972). The original analyses assumed an internal thermal load of 11.5 kW and the NAC-1 SAR assumed an internal 750 W thermal load. These analyses considered the NAC-1 cask with and without the ISO shipping container and assumed bounding conditions for the fire accident scenario, which has a maximum internal thermal load of 405 W. Examination of the previous fire accident has shown that these temperature limits are not exceeded for an internal thermal load of 11.5 kW or 750 W. The maximum temperature at the center of the fuel during a hypothetical fire accident is 589.8 °F (SNF-4794). The 644 °F temperature limit for the cladding is not exceeded anywhere in the array of stacked rods. The transportation SAR and the NAC-1 SAR analyses found that the integrity of the NAC-1 cask was not lost for the design fire accident conditions and consequently, no credible radioactive releases occur for this event.

D3.4.2.7.2 Source Term Analysis. The SNF storage systems will not release radioactive material and fuel cladding temperature limits are not exceeded; therefore, a source term analysis is not required.

D3.4.2.7.3 Consequence Analysis.

Fast Flux Test Facility

The fire event does not result in a condition where the storage casks release radioactive material; therefore, a consequence analysis is not needed for this event.
NRF TRIGA

The fire event does not result in a condition where the storage casks release radioactive material; therefore, a consequence analysis has not been developed.

Light Water Reactor Fuel

The fire analyses do not result in a scenario that causes the release of radioactive material; therefore, a consequence analysis has not been developed.

D3.4.2.7.4 Comparison to Guidelines.

Unmitigated event – For the hypothetical accidents described in Section D3.4.1, the bounding consequences for each fuel type show that the off-site release limits are not challenged. Therefore, no safety-class confinement SSCs are necessary.

For the NRF TRIGA fuel, the on-site consequences do not exceed the on-site risk evaluation guidelines. Therefore, for the purposes of controlling accident releases following a fire accident, no safety-significant SSCs are required to confine the NRF TRIGA fuel. However, the NRF TRIGA casks do provide structural protection of the SNF payload, and as such, are designated as important-to-safety Category B for NRC equivalency. On this basis, the NRF TRIGA casks are designated as Safety Significant.

For FFTF fuel and the LWR fuel in NAC-1 casks, the bounding on-site consequences for the hypothetical accident analyzed exceed the on-site risk evaluation guidelines. Therefore, safety-significant SSCs are required to prevent or mitigate the dose consequences.

While no safety-class SSCs are required to provide confinement, some SSCs are required to protect the geometric configuration of the fuel for criticality safety. SSCs credited with retaining the critically safe geometry of the fuel are designated as Safety Class. Temperatures from the design basis fire were evaluated to demonstrate the materials of construction can continue to perform their safety function.

Mitigated event – For the fire events postulated and analyzed in the scenario development, the potential releases are prevented. These events take credit for the packaging components providing confinement of the fuel for the analyzed fire events.

D3.4.2.7.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls. The following SSCs and TSR controls ensure that the SNF is adequately confined and protected from the fire events within design values.
The following TSR control is necessary to protect the assumptions made in the fire analyses:

- Fire loadings are to be controlled per the fire hazard analysis.

**Fast Flux Test Facility Fuel**

The following SSCs are designated as Safety Significant or Safety Class:

- **CCC** – The CCC is designated as Safety Class to maintain structural integrity for criticality geometry control.

- **FFTF ISC** – The ISC is designated as Safety Significant to withstand transportation design basis fire conditions without the loss of confinement or exceeding the temperature limits for fuel cladding or cask components, as well as to provide structural integrity protection.

**NRF TRIGA**

The following SSCs are designated as Safety Significant:

- **DOT-6M container** – The DOT-6M container is designated as Safety Significant for structural integrity protection.

- **2R container** – The 2R container is designated as Safety Significant for structural integrity protection.

- **NRF TRIGA cask** – The TRIGA casks are designated as Safety Significant for NRC important-to-safety considerations to provide structural integrity protection against the fire event.

- **Rad-Vault** – The Rad-Vault is designated as Safety Significant for structural integrity protection.

**Light Water Reactor Fuel**

The following SSCs are designated as Safety Significant or Safety Class:

- **LWR canister** – The LWR canister is designated as Safety Class to maintain structural integrity for criticality geometry control. In addition, the LWR canister is designated as Safety Significant for confinement.
• **NAC-1 cask system** – The NAC-1 cask is designated as Safety Significant to provide structural integrity protection against the fire event.

• **ISO container** – The ISO container is designated as Safety Significant for structural integrity protection.

### D3.4.3 Beyond Design Basis Accidents

DOE Order 5480.23 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 design basis accidents (BDBAs) are not analyzed to the same level of detail as DBAs, but do provide insight into the magnitude of consequences of BDBAs. This insight from BDBA analysis has the potential for identifying additional facility features that could prevent or reduce severe BDBA consequences. While these events would be beyond requirements of further safety-class or safety-significant functions, they might provide guidance to the prioritization of long-term safety improvements for a facility. The Order specifically excludes evaluation of human-generated external events as BDBAs.

Hypothetical release source terms are developed for each fuel type in Section D3.4.1.1 for an unmitigated release from each fuel type cask to determine the classification of engineered barriers that prevent an uncontrolled release. The scenarios are hypothetical and non-mechanistic. These hypothetical scenarios also establish the magnitude of a release from a BDBA.

Given the extremely low likelihoods associated with such BDBAs and the calculated consequences for the hypothetical scenarios in Section D3.4.1.1, no new insights into facility operations are expected.

### D3.5 REFERENCES


ACI-349, 1990, *Code Requirements for Nuclear Safety Related Concrete Structures,* American Concrete Institute, Farmington Hills, Michigan.


CHAPTER D4.0

SAFETY STRUCTURES, SYSTEMS, AND COMPONENTS
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D4.0 SAFETY STRUCTURES, SYSTEMS, AND COMPONENTS

D4.1 INTRODUCTION

This chapter provides details of the structures, systems, and components (SSCs) that are necessary for the facility to meet off-site release limits, satisfy on-site risk guidelines, provide significant defense in depth, or contribute to worker safety. This chapter meets the requirements of U.S. Department of Energy (DOE) Order 5480.23, Nuclear Safety Analysis Reports, follows the format guidance of DOE-STD-3009-94, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports, and includes the following information:

- A list of safety-class SSCs and subsections containing safety function details, a system description, functional requirements, a system evaluation, and a list of assumptions requiring Technical Safety Requirements (TSRs) to ensure performance of the safety function.
- A list of the safety-significant SSCs and subsections containing safety function details, a system description, functional requirements, a system evaluation, and a list of assumptions requiring TSRs to ensure performance of the safety function.

In general, safety-class SSCs are those items required for protection of the off-site environment and the public, or the prevention of an inadvertent criticality. Safety-class SSCs include items designated as "Safety Class" in accordance with DOE Order 6430.1A, General Design Criteria. Safety-class SSCs also encompass items that are designated as "important to safety" and have been classified as Category A, as defined in Item 29 of HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward Additional NRC Requirements.

Safety-significant items are those SSCs required for the protection of on-site personnel not directly involved in facility operations. Safety-significant SSCs also encompass items that have been designated important-to-safety Category B in accordance with Item 29 of HNF-SD-SNF-DB-003. SSCs that would prevent or mitigate an on-site fatality or protect large numbers of facility workers (not industrial safety), or those SSCs that would prevent or mitigate toxic chemical exposures are also designated Safety Significant.

All SSCs that are not classified as Safety Class or Safety Significant are general-service SSCs. The SSCs that have been designated as important-to-safety Category C are also included in general-service SSCs. General-service SSCs protect workers from standard industrial hazards or are controlled by Site safety programs.

The majority of the equipment in the 200 Area Interim Storage Area (ISA) is classified as General Service (excluding the dry cask storage systems). The dry cask storage systems have
been classified as Safety Significant. The core component container (CCC) for Fast Flux Test Facility (FFTF) fuel and the light water reactor (LWR) canister for commercial LWR fuel have been classified as Safety Class. This classification was based on NUREG/CR-6407, Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety, and DOE 6430.1A, Section 1300-4, because of their credited function for criticality prevention.

The accident analyses in Chapter D3.0 identified one or more design basis accidents (DBAs) whose unmitigated consequences bound all accident sequences and scenarios identified for that category in the hazard analysis. The process identified criticality-related safety-class considerations requiring that the FFTF interim storage cask (ISC) CCC and the LWR canister provide criticality geometry control of the stored fuel. The analysis also identified safety-significant considerations for components of the cask storage systems that mitigated potential consequences in excess of on-site guidelines. No events were identified that had off-site consequences in excess of allowable limits.

Each accident analysis section in Chapter D3.0 concludes with a summary of safety SSCs that provides the basis for this chapter. A summary of the accident categories and the designated safety SSCs that prevent or mitigate their consequences is provided in Table D4-1. Table D4-2 provides a summary of the safety SSCs identified and provides the level of safety credited for each accident category. Many SSCs provide a safety function for prevention or mitigation of more than one accident. Detailed definitions of safety-class and safety-significant SSCs are provided in Sections D4.3 and D4.4.

A probabilistic risk analysis was performed to determine the frequency of a tornado wind-borne missile striking any of the dry cask storage systems planned for the 200 Area ISA (HNF-1785). The analysis concludes that the rate at which storage area safety is compromised is less than $10^{-8}$ per year. Generation of tornado-based, wind-borne missiles that could compromise cask system integrity is therefore not credible and evaluation is not required.
<table>
<thead>
<tr>
<th>Design Basis Accident (Chapter D3.0)</th>
<th>Safety structures, systems and components&lt;sup&gt;60&lt;/sup&gt;</th>
<th>Safety function</th>
<th>Summary justification of safety function</th>
<th>Controls</th>
</tr>
</thead>
<tbody>
<tr>
<td>Handling/Drop</td>
<td>FFTF Interim Storage Cask</td>
<td>Maintain confinement of radioactive materials after a credible drop and provide passive protection of the CCC such that it retains structural integrity.</td>
<td>Design requirement of the ISC is to maintain confinement after credible drops. The CCC inside the ISC provides support to the fuel assemblies and maintains geometry control for criticality prevention. Structural analysis demonstrates integrity of the CCC.</td>
<td>AC 5.9 Lift Restrictions – lift height restriction.</td>
</tr>
<tr>
<td></td>
<td>CCC</td>
<td>Maintain structural integrity within the ISC to provide criticality geometry control after credible drops.</td>
<td>The CCC inside the ISC provides support to the fuel assemblies and maintains geometry control for criticality prevention. Structural analysis, including the buckling analysis, demonstrates integrity of the CCC.</td>
<td>AC 5.9 Lift Restrictions – lift height restriction.</td>
</tr>
<tr>
<td></td>
<td>NRF TRIGA Cask</td>
<td>Maintain confinement of radioactive materials after a credible drop.</td>
<td>Design requirement of the cask is to maintain confinement after credible drops.</td>
<td>AC 5.9 Lift Restrictions – lift height restriction.</td>
</tr>
<tr>
<td></td>
<td>DOT-6M Container</td>
<td>Provide impact absorption for the 2R container.</td>
<td>Design requirement of the DOT-6M is to provide impact absorption so the 2R container can maintain confinement after credible drops.</td>
<td>AC 5.9 Lift Restrictions – lift height restriction.</td>
</tr>
<tr>
<td></td>
<td>2R Container</td>
<td>Maintain confinement of radioactive materials after a credible drop within the DOT-6M.</td>
<td>Design requirement of the 2R container is to maintain confinement after credible drops.</td>
<td>AC 5.9 Lift Restrictions – lift height restriction.</td>
</tr>
<tr>
<td></td>
<td>NAC-1 Cask</td>
<td>Provide passive protection of the LWR canister such that it retains structural integrity after a credible drop.</td>
<td>Design requirement of the NAC-1 cask is to provide impact absorption so the LWR canister can maintain confinement after credible drops. The LWR canister design provides support to the fuel assembly or pins and maintains geometry control for criticality prevention.</td>
<td>AC 5.9 Lift Restrictions – lift height restriction.</td>
</tr>
<tr>
<td>Design Basis Accident (Chapter D3.0)</td>
<td>Safety structures, systems and components (1)</td>
<td>Safety function</td>
<td>Summary justification of safety function</td>
<td>Controls</td>
</tr>
<tr>
<td>-------------------------------------</td>
<td>-----------------------------------------------</td>
<td>-----------------</td>
<td>----------------------------------------</td>
<td>----------</td>
</tr>
<tr>
<td>Handling/Drop (continued)</td>
<td>LWR Canister</td>
<td>Maintain confinement of radioactive materials after a credible drop (SS) within NAC-1 cask and maintain structural integrity to provide criticality geometry control (SC).</td>
<td>The LWR canister maintains confinement after credible drops. The canister design provides support to the fuel assembly or pins and maintains geometry control for criticality prevention. Structural analysis including buckling analysis demonstrates integrity of the LWR canister.</td>
<td>AC 5.9 Lift Restrictions - lift height restriction.</td>
</tr>
<tr>
<td>Mobile Crane Fall</td>
<td>FFTF Interim Storage Cask</td>
<td>Maintain structural integrity sufficient to maintain confinement of radioactive materials after the crane fall.</td>
<td>The ISC is analyzed to maintain confinement after a crane fall accident.</td>
<td>AC 5.11 Crane Utilization.</td>
</tr>
<tr>
<td>NAC-1 Cask System</td>
<td>X</td>
<td>Provide passive protection of the LWR canister such that it maintains confinement after a crane fall accident.</td>
<td>The NAC-1 cask is analyzed to provide impact absorption so the LWR canister can maintain confinement after credible drops.</td>
<td>AC 5.11 Crane Utilization.</td>
</tr>
<tr>
<td>Cask Tipover</td>
<td>FFTF Interim Storage Cask System</td>
<td>Maintain confinement of radioactive materials after cask tipover and provide passive protection of the CCC such that it retains structural integrity.</td>
<td>Design requirement of the ISC is to maintain confinement after cask tipover. The CCC inside the ISC provides support to the fuel assemblies and maintains geometry control for criticality prevention. Structural analysis demonstrates integrity of the CCC.</td>
<td>AC 5.10 Spacing and Placement.</td>
</tr>
<tr>
<td>CCC</td>
<td>X</td>
<td>Maintain structural integrity to provide criticality geometry control.</td>
<td>The CCC inside the ISC provides support to the fuel assemblies and maintains geometry control for criticality prevention. Structural analysis, including the buckling analysis, demonstrates integrity of the CCC.</td>
<td>(Design feature)</td>
</tr>
<tr>
<td>Rad-Vault</td>
<td>X</td>
<td>Passive design features preclude tipping.</td>
<td>The Rad-Vault has been analyzed to not tip over or slide as a result of NPH or DBAs.</td>
<td>AC 5.10 Spacing and Placement. (Design feature)</td>
</tr>
<tr>
<td>Safety function</td>
<td>Summary justification of safety function</td>
<td>Controls</td>
<td>Design feature</td>
<td></td>
</tr>
<tr>
<td>-----------------</td>
<td>----------------------------------------</td>
<td>----------</td>
<td>---------------</td>
<td></td>
</tr>
<tr>
<td>Maintain confinement of radioactive materials after rupture of all fuel pins</td>
<td>Design requirement of the TRIGA cask is to maintain confinement after rupture of all fuel pins by not exceeding the design pressure.</td>
<td>(Design feature)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Maintain structural integrity to provide criticality geometry, control at the pressure generated from the rupture of all fuel pins.</td>
<td></td>
<td>(Design feature)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Maintain confinement of radioactive materials after rupture of all fuel pins.</td>
<td></td>
<td>(Design feature)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Maintain confinement of radioactive materials after rupture of all fuel pins.</td>
<td></td>
<td>(Design feature)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Passive design features preclude container tipping</td>
<td>Design requirement of the CCC is to structurally withstand the pressure after rupture of all fuel pins by not exceeding the design pressure.</td>
<td>(Design feature)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Maintain confinement of radioactive materials after rupture of all fuel pins.</td>
<td>Design requirement of the NAC-I cask in the ISO container has been analyzed to not tip over or slide as a result of NPII or DBAs.</td>
<td>(Design feature)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Maintain confinement of radioactive materials after rupture of all fuel pins.</td>
<td></td>
<td>(Design feature)</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>SC</th>
<th>SS</th>
<th>Description</th>
<th>Safety structures, systems and components</th>
</tr>
</thead>
<tbody>
<tr>
<td>X</td>
<td>X</td>
<td>NRC TRIGA Cask in DOT-6M Container</td>
<td>2R Container</td>
</tr>
<tr>
<td>X</td>
<td>X</td>
<td>NAC-I Cask in ISO Container</td>
<td>FFTE Interim Storage Cask</td>
</tr>
<tr>
<td>X</td>
<td>X</td>
<td>CCC</td>
<td>(NRC TRIGA Cask)</td>
</tr>
<tr>
<td>X</td>
<td>X</td>
<td>2R Container</td>
<td>LWR Canister</td>
</tr>
</tbody>
</table>

**Table D4-1. Safety-Class and Safety-Significant Structures, Systems and Components Summary**
<table>
<thead>
<tr>
<th>Design Basis Accident (Chapter D3.0)</th>
<th>Safety structures, systems and components(*)</th>
<th>Safety function</th>
<th>Summary justification of safety function</th>
<th>Controls</th>
</tr>
</thead>
<tbody>
<tr>
<td>Seismic</td>
<td>FFTF Interim Storage Cask</td>
<td>X Withstand seismic accelerations without loss of structural integrity sufficient to maintain confinement and without tipover or sliding.</td>
<td>Although the ISC was analyzed to survive a cask tipover, seismic analysis precludes common mode failure involving multiple casks.</td>
<td>AC 5.10 Spacing and Placement.</td>
</tr>
<tr>
<td>CCC</td>
<td>X Withstand seismic accelerations to maintain structural integrity to provide criticality geometry control.</td>
<td>All seismic accelerations are bounded by the drop analysis.</td>
<td>(Design feature)</td>
<td></td>
</tr>
<tr>
<td>Rad-Vault</td>
<td>X Withstand seismic accelerations without tipover or sliding.</td>
<td>Seismic analysis precludes common mode failure involving multiple casks and supplants requirement to perform cask tipover analysis.</td>
<td>AC 5.10 Spacing and Placement.</td>
<td></td>
</tr>
<tr>
<td>NRF TRIGA Cask</td>
<td>X Withstand seismic accelerations without loss of confinement.</td>
<td>All seismic accelerations are bounded by drop analysis.</td>
<td>(Design feature)</td>
<td></td>
</tr>
<tr>
<td>DOT-6M</td>
<td>X Provide impact absorption for the 2R container.</td>
<td>Design requirement of the DOT-6M is to provide impact absorption so the 2R container can maintain confinement after credible drops.</td>
<td>(Design feature)</td>
<td></td>
</tr>
<tr>
<td>2R Container</td>
<td>X Withstand seismic accelerations without loss of confinement.</td>
<td>Seismic accelerations are bounded by the drop analysis.</td>
<td>(Design feature)</td>
<td></td>
</tr>
<tr>
<td>LWR Canister</td>
<td>X Withstand seismic accelerations without loss of confinement (SS) and maintain structural integrity to provide criticality geometry control (SC).</td>
<td>All seismic accelerations are bounded by drop analysis.</td>
<td>(Design feature)</td>
<td></td>
</tr>
<tr>
<td>NAC-1 Cask in ISO Container</td>
<td>X Provide structural support to the LWR canister to withstand seismic accelerations without tipover or sliding.</td>
<td>The NAC-1 cask in the ISO container has been analyzed not to tip over or slide as a result of a seismic event.</td>
<td>AC 5.10 Spacing and Placement. (Design feature)</td>
<td></td>
</tr>
<tr>
<td>Design Basis Accident (Chapter D3.0)</td>
<td>Safety structures, systems and components(^{(1)})</td>
<td>Safety function</td>
<td>Summary justification of safety function</td>
<td>Controls</td>
</tr>
<tr>
<td>-------------------------------------</td>
<td>--------------------------------------------------</td>
<td>----------------</td>
<td>------------------------------------------</td>
<td>----------</td>
</tr>
<tr>
<td>Tornado/Wind</td>
<td>FFTF Interim Storage Cask</td>
<td>X Withstand DBT winds (excluding DBT missiles) without sliding or tip over. Also withstand the design basis wind and wind-driven missiles established for the 200 Area ISA without loss of structural integrity sufficient to cause loss of confinement.</td>
<td>Tornado analysis precludes common mode failure involving multiple casks, and the wind missile analysis demonstrates no loss of confinement and environmental protection provided to the CCC.</td>
<td>AC 5.10 Spacing and Placement.</td>
</tr>
<tr>
<td>Rad-Vault</td>
<td>X Withstand DBT winds (excluding DBT missiles) without sliding or tip over. Also withstand the design basis wind and wind-driven missiles established for the 200 Area ISA.</td>
<td>Tornado analysis precludes common mode failure involving multiple casks, and the wind missile analysis demonstrates environmental protection provided to TRIGA casks and DOT-6M/2R containers.</td>
<td>AC 5.10 Spacing and Placement.</td>
<td></td>
</tr>
<tr>
<td>NRF TRIGA Cask</td>
<td>X Withstand tornado pressure differential without loss of confinement.</td>
<td>The design pressure is much greater than the 0.9 lbs/in(^2) gauge tornado pressure differential.</td>
<td>(Design feature)</td>
<td></td>
</tr>
<tr>
<td>2R Container</td>
<td>X Withstand tornado pressure differential without loss of confinement.</td>
<td>The design pressure is much greater than the 0.9 lbs/in(^2) gauge tornado pressure differential.</td>
<td>(Design feature)</td>
<td></td>
</tr>
<tr>
<td>NAC-1 Cask</td>
<td>X Withstand DBT winds (excluding DBT missiles). Also withstand the design basis wind and wind-driven missiles established for the 200 Area ISA, without structural damage to the LWR canister.</td>
<td>Tornado analysis precludes common mode failure involving tipover of multiple casks, and the wind-missile analysis demonstrates environmental protection provided to the LWR canister.</td>
<td>AC 5.10 Spacing and Placement.</td>
<td></td>
</tr>
<tr>
<td>LWR Canister</td>
<td>X Withstand tornado pressure differential without loss of confinement.</td>
<td>The design pressure is much greater than the 0.9 lbs/in(^2) gauge tornado pressure differential.</td>
<td>(Design feature)</td>
<td></td>
</tr>
<tr>
<td>ISO Container</td>
<td>X Provide structural support to the NAC-1 cask to withstand tornado winds without tipover or sliding.</td>
<td>The NAC-1 cask in the ISO container has been analyzed not to tip over or slide as a result of a tornado event.</td>
<td>AC 5.10 Spacing and Placement.</td>
<td></td>
</tr>
<tr>
<td>Design Basis Accident (Chapter D3.0)</td>
<td>Safety structures, systems and components</td>
<td>Safety function</td>
<td>Summary justification of safety function</td>
<td>Controls</td>
</tr>
<tr>
<td>--------------------------------------</td>
<td>------------------------------------------</td>
<td>----------------</td>
<td>--------------------------------------</td>
<td>----------</td>
</tr>
<tr>
<td>Fire</td>
<td>FFTF Interim Storage Cask</td>
<td>X</td>
<td>Withstand transportation DBF conditions without loss of structural integrity sufficient to cause loss of confinement or exceeding temperature limits for fuel cladding or cask components.</td>
<td>ISC design precludes exceeding temperature limits for fuel cladding or cask components and maintains confinement during a DBF. Precludes common mode failure involving multiple casks. AC 5.12 Combustible Loading Limits.</td>
</tr>
<tr>
<td>CCC</td>
<td>X</td>
<td>Withstand transportation DBF conditions inside ISC without exceeding temperature limits for fuel cladding or container components to maintain structural integrity for criticality geometry control.</td>
<td>CCC can withstand temperatures inside ISC during a DBF such that temperatures do not exceed limits, maintaining structural integrity.</td>
<td>AC 5.12 Combustible Loading Limits.</td>
</tr>
<tr>
<td>Rad-Vault</td>
<td>X</td>
<td>Withstand transportation DBF conditions such that TRIGA cask and DOT-6M/2R containers inside do not lose confinement or exceed temperature limits for fuel cladding or cask components.</td>
<td>Rad-Vault provides environmental protection of the TRIGA casks and DOT-6M/2R containers during a DBF such that their confinement function is not breached and temperatures do not exceed limits. Precludes common mode failure involving multiple casks.</td>
<td>AC 5.12 Combustible Loading Limits.</td>
</tr>
<tr>
<td>NRF TRIGA Cask</td>
<td>X</td>
<td>Withstand transportation DBF conditions within the Rad-Vault without losing confinement or exceeding temperature limits for fuel cladding or cask components.</td>
<td>TRIGA casks can withstand temperatures inside the Rad-Vault during a DBF such that their confinement function is not breached and temperatures do not exceed limits.</td>
<td>AC 5.12 Combustible Loading Limits.</td>
</tr>
<tr>
<td>DOT-6M</td>
<td>X</td>
<td>Withstand transportation DBF conditions within the Rad-Vault such that the 2R container inside does not lose confinement or exceed temperature limits for fuel cladding or container components.</td>
<td>DOT-6M containers provide thermal insulation during a DBF such that the 2R container inside does not lose its confinement function and temperatures do not exceed limits.</td>
<td>AC 5.12 Combustible Loading Limits.</td>
</tr>
</tbody>
</table>
Table D4-1. Safety-Class and Safety-Significant Structures, Systems and Components Summary. (8 sheets)

<table>
<thead>
<tr>
<th>Design Basis Accident (Chapter D3.0)</th>
<th>Safety structures, systems and components⁽¹⁾</th>
<th>Safety function</th>
<th>Summary justification of safety function</th>
<th>Controls</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fire (continued)</td>
<td>2R Container</td>
<td>X</td>
<td>Withstand transportation DBF conditions within the Rad-Vault without losing confinement or exceeding temperature limits for fuel cladding or container components.</td>
<td>2R containers can withstand temperatures inside the Rad-Vault during a DBF such that their confinement function is not breached and temperatures do not exceed limits.</td>
</tr>
<tr>
<td></td>
<td>NAC-1 Cask</td>
<td>X</td>
<td>Withstand transportation DBF conditions such that the LWR canister does not lose confinement or exceed temperature limits for fuel cladding or cask components.</td>
<td>NAC-1 cask provides environmental protection of the LWR canisters during a DBF such that their confinement function is not breached and temperatures do not exceed limits. Precludes common mode failure involving multiple casks.</td>
</tr>
<tr>
<td></td>
<td>LWR Canister</td>
<td>X</td>
<td>Withstand transportation DBF conditions inside the NAC-1 cask without loss of confinement (SS) or exceeding temperature limits for fuel cladding or container components to maintain structural integrity for criticality geometry control (SC).</td>
<td>LWR canisters can withstand temperatures inside the NAC-1 cask during a DBF such that confinement is not breached and temperatures do not exceed limits, maintaining structural integrity.</td>
</tr>
</tbody>
</table>
### Table D4-1. Safety-Class and Safety-Significant Structures, Systems and Components Summary. (8 sheets)

<table>
<thead>
<tr>
<th>Design Basis Accident (Chapter D3.0)</th>
<th>Safety structures, systems and components&lt;sup&gt;(1)&lt;/sup&gt;</th>
<th>Safety function</th>
<th>Summary justification of safety function</th>
<th>Controls</th>
</tr>
</thead>
<tbody>
<tr>
<td>Criticality</td>
<td>FFTF CCC</td>
<td>X Maintain structural integrity to provide criticality geometry control under all credible normal and accident conditions using double contingency principles and $K_{en} \leq 0.95$.</td>
<td>Dimensional parameters credited in the CSERs must be maintained under all credible normal and accident conditions by retaining structural integrity of credited components, as demonstrated in the structural analyses, including the buckling analysis.</td>
<td>AC 5.7 Nuclear Criticality Safety. AC 5.9 Lift Restrictions.</td>
</tr>
</tbody>
</table>
| LWR Canister                         | X Maintain structural integrity to provide criticality geometry control under all credible normal and accident conditions using double contingency principles and $K_{en} \leq 0.95$. | Dimensional parameters credited in the CSERs must be maintained under all credible normal and accident conditions by retaining structural integrity of credited components, as demonstrated in the structural analyses, including the buckling analysis. | AC 5.7 Nuclear Criticality Safety. AC 5.9 Lift Restrictions. |}

<sup>(1)</sup> Safety function identifies safety-significant or safety-class function required to prevent or mitigate the specific accident.

**Abbreviations:**
- CCC = core component container.
- CSER = criticality safety evaluation report.
- DBA = design basis accident.
- DBF = design basis fire.
- DBT = design basis tornado.
- DOT = U.S. Department of Transportation.
- FFTF = Fast Flux Test Facility.
- GS = general service.
- ISC = interim storage cask.
- ISO = International Standards Organization.
- NA = not applicable.
- NAC = Nuclear Assurance Corporation.
- NPH = natural phenomena hazard.
- NRF = Neutron Radiography Facility.
- SC = safety class.
- SS = safety significant.
- TRIGA = Training, Research and Isotope Production, General Atomics.
<table>
<thead>
<tr>
<th>Structures, systems, and components</th>
<th>Handling/drop accident</th>
<th>Mobile crane fall</th>
<th>Cask tipover</th>
<th>Fuel rod rupture</th>
<th>Seismic</th>
<th>Tornado/wind</th>
<th>Fire</th>
<th>Confinement boundary</th>
<th>Criticality prevention</th>
<th>NRC important to safety classification</th>
<th>FSAR section</th>
</tr>
</thead>
<tbody>
<tr>
<td>FFTF ISC</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
<td>X</td>
<td></td>
<td>ITS-Cat B</td>
<td>D4.4.1</td>
</tr>
<tr>
<td>CCC</td>
<td>SC(1)</td>
<td>SC(1)</td>
<td>SC(1)</td>
<td>SC(1)</td>
<td>SC(1)</td>
<td></td>
<td>X</td>
<td></td>
<td></td>
<td>ITS-Cat A</td>
<td>D4.3.1</td>
</tr>
<tr>
<td>Rad-Vault</td>
<td>SS(2)</td>
<td>SS(2)</td>
<td>SS(2)</td>
<td>SS(2)</td>
<td>SS(2)</td>
<td></td>
<td>X</td>
<td></td>
<td></td>
<td>ITS-Cat B</td>
<td>D4.4.2</td>
</tr>
<tr>
<td>NRF TRIGA Cask</td>
<td>SS(2)</td>
<td>SS(2)</td>
<td>SS(2)</td>
<td>SS(3)</td>
<td>SS(3)</td>
<td>X</td>
<td>X</td>
<td></td>
<td></td>
<td>ITS-Cat B</td>
<td>D4.4.3</td>
</tr>
<tr>
<td>DOT-6M</td>
<td>SS(2)</td>
<td>SS(2)</td>
<td>SS(3)</td>
<td>SS(5)</td>
<td>SS(5)</td>
<td></td>
<td>X</td>
<td></td>
<td></td>
<td>ITS-Cat B</td>
<td>D4.4.4</td>
</tr>
<tr>
<td>2R Container</td>
<td>SS(2)</td>
<td>SS(3)</td>
<td>SS(2)</td>
<td>SS(5)</td>
<td>SS(5)</td>
<td></td>
<td>X</td>
<td></td>
<td></td>
<td>ITS-Cat B</td>
<td>D4.4.4</td>
</tr>
<tr>
<td>NAC-1</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
<td></td>
<td>X</td>
<td></td>
<td></td>
<td>ITS-Cat B</td>
<td>D4.4.5</td>
</tr>
<tr>
<td>LWR Canister</td>
<td>SS/SC(1)</td>
<td>SS/SC(1)</td>
<td>SS/SC(1)</td>
<td>SS</td>
<td>SS/SC(5)</td>
<td>X</td>
<td>X</td>
<td></td>
<td></td>
<td>ITS-Cat A</td>
<td>D4.3.2</td>
</tr>
<tr>
<td>ISO Container</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
<td>SS</td>
<td></td>
<td>X</td>
<td></td>
<td></td>
<td>ITS-Cat B</td>
<td>D4.4.5</td>
</tr>
<tr>
<td>Crane</td>
<td>GS(3)</td>
<td>GS(2)</td>
<td>GS(2)</td>
<td>GS(2)</td>
<td>GS(2)</td>
<td></td>
<td>X</td>
<td></td>
<td></td>
<td>ITS-Cat B</td>
<td>NA</td>
</tr>
<tr>
<td>Lifting rigging</td>
<td>GS(3)</td>
<td>GS(3)</td>
<td>GS(3)</td>
<td>GS(3)</td>
<td>GS(3)</td>
<td></td>
<td>X</td>
<td></td>
<td></td>
<td>ITS-Cat B</td>
<td>NA</td>
</tr>
</tbody>
</table>

(1) This item is designated as Safety Class for structural integrity to provide criticality geometry control.
(2) This item upgraded to Safety Significant to accomplish NRC equivalency based on important-to-safety classification and due to its significant contribution to defense-in-depth and worker safety.
(3) Critical lift requirements imposed using DOE/RL-92-36, Hanford Site Hoisting and Rigging Manual, accomplish NRC equivalency for important-to-safety Category B.

<table>
<thead>
<tr>
<th>CCC</th>
<th>DOT</th>
<th>FFTF</th>
<th>FSAR</th>
<th>GS</th>
<th>ISC</th>
<th>ISO</th>
<th>ITS-Cat</th>
</tr>
</thead>
<tbody>
<tr>
<td>NA</td>
<td>NAC</td>
<td>NRC</td>
<td>NRF</td>
<td>SC</td>
<td>SS</td>
<td>TRIGA</td>
<td>not applicable.</td>
</tr>
</tbody>
</table>
D4.2 REQUIREMENTS

This section identifies design codes, standards, regulations, and DOE Orders that are required for establishing the facility safety basis. The intent is to provide only the requirements that are specific to this chapter and pertinent to the safety basis. Specific codes, standards, and requirements applicable to the 200 Area ISA are defined in the Spent Nuclear Fuel (SNF) Project Standards/Requirements Identification Document (HNF-SD-SNF-RD-001).

The following DOE Orders, regulations, and standards are applicable to the safety basis for the facility:

- **Title 10, Code of Federal Regulations (CFR), Section 830.120, "Quality Assurance Requirements."** This rule requires that a sufficient quality assurance program be in place.

- **10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."** This rule is used for the licensing of independent spent fuel storage installations. 10 CFR 72.3, "Definitions," defines SSCs that are considered "important to safety."

- **10 CFR 835, "Occupational Radiation Protection."** This rule provides requirements for radiation protection programs.

- **DOE Order 5480.22, Technical Safety Requirements.** This order sets the requirements for the development and preparation of a TSR document, which is prepared separately.

- **DOE Order 5480.23, Nuclear Safety Analysis Reports.** This order provides nuclear safety analysis report (SAR) content requirements.

- **DOE Order 6430.1A, General Design Criteria.** This order provides general criteria and guidance for facility and system design. In addition, Division 13, "Special Facilities," Section 1300, "General Requirements," and Section 1320, "Irradiated Fissile Material Storage Facilities," requirements are imposed for the 200 Area ISA. Compliance with DOE Order 6430.1A is demonstrated in SNF-5139, **DOE 6430.1A Compliance Analysis for the 200 Area Interim Storage Area.**

- **DOE/RL-92-36, Hanford Site Hoisting and Rigging Manual.** This manual provides the requirements for lifting and rigging equipment and material.
The following standards are used for content and guidance:

- **DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports.* This standard is used to determine hazard categories for nuclear facilities.

- **DOE Order 5480.28, *Natural Phenomena Hazards Mitigation.* This order is used to define design requirements for seismic events and straight wind.

- **DOE-STD-1020-94, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities.* This standard provides natural phenomena (NPH) design and evaluation criteria. The 200 Area ISA was designed and evaluated for seismic events and straight wind in accordance with this standard.

- **DOE-STD-1021-93, *Natural Phenomena Hazards Performance Categorization Guidelines for Structures, Systems, and Components.* This standard is used to define the specific performance category for SSCs.

- **DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports.* This standard supplements DOE Order 5480.23 by providing guidance specific to nonreactor nuclear facilities. In this regard, the standard provides more detailed information on the performance of accident analyses for Hazard Category 2 and 3 facilities. The standard also establishes additional requirements for the establishment of defense in depth and the identification of safety-significant SSCs.

In Letter 95-SFD-167, *Implementation of K Basins Spent Nuclear Fuel Project (SNFP) Regulatory Policy* (Sellers 1995), DOE established the requirement for new SNF Project facilities to achieve "nuclear safety equivalency" to comparable U.S. Nuclear Regulatory Commission (NRC)-licensed facilities. The SNF Project identified the NRC requirements that were needed in addition to existing applicable DOE requirements to establish nuclear safety equivalency. These NRC requirements and the process used to identify them are documented in HNF-SD-SNF-DB-003 and WHC-SD-SNF-DB-009, *Canister Storage Building Natural Phenomena Hazards,* Appendix C, "200 East Area Interim Storage Area Natural Phenomena Hazards." The NRC requirements are used for licensing independent spent fuel storage installations. 10 CFR 72.3, "Definitions," defines SSCs that are considered important to safety. 10 CFR 72.122, "Overall Requirements," requires that the design bases for SSCs important to safety reflect appropriate combinations of effects of normal and accident conditions and the effects of natural phenomena.

For the 200 Area ISA, important-to-safety SSCs have been identified in accordance with 10 CFR 72.3. Once SSCs have been identified as having a function meeting the definition of important to safety, the requirements for SSCs important to safety specified in 10 CFR 72 are
imposed. A graded approach is applied to an SSC important to safety by using the guidance provided in NUREG/CR-6407, as follows:

- **Category A – Critical to Safe Operation**

  SSCs in this category include those whose failure or malfunction could directly result in a condition adverse to public health and safety. Important-to-safety SSCs in this category are classified as Safety Class, as defined in DOE Order 6430.1A with the additional requirements therein.

- **Category B – Major Impact on Safety**

  SSCs in this category include those whose failure or malfunction could result in a condition adversely affecting collocated worker health and safety. Note that from the definition of Category C, Category B is understood to include events that could significantly damage the storage containers without severe impact to public health and safety. SSCs in this category are classified as Safety Significant.

- **Category C – Minor Impact on Safety**

  SSCs whose failure or malfunction would not significantly reduce containment and would not be likely to create a situation adversely affecting public or collocated workers’ health and safety. SSCs in this category are classified as General Service.

In conjunction with the requirements noted above, ANSI/ANS-57.9-1992, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)*, provides guidance on design and performance criteria specific to a dry storage concept. These criteria are intended to be consistent with meeting the 10 CFR 72 requirements.

Additional discussion of the NRC criteria for mitigation of natural phenomena is provided in Section B1.2 of the SNF Project Final Safety Analysis Report (FSAR), Annex B. Specific NPH design requirements implementing NRC equivalency requirements have been established for the 200 ISA in WHC-SD-SNF-DB-009, Appendix C.

Letter 97-SFD-172, *Risk Evaluation Guidelines (REGs) to Ensure Inherently Safer Designs* (Sellers 1997), provides SNF Project guidance for design criteria with regard to exposure limits to be used in evaluating risk to the public and to on-site personnel.

These documents establish the design requirements for the 200 Area ISA.
D4.3 SAFETY-CLASS STRUCTURES, SYSTEMS, AND COMPONENTS

Safety-class items are SSCs, including portions of storage systems, whose failure could adversely affect the environment or the safety and health of the public. Detailed definitions of safety-class SSCs are provided in Section D4.2.

The SSCs credited with a safety-class function in Chapter D3.0 are described in the subsections that follow.

D4.3.1 Core Component Container for Fast Flux Test Facility Spent Fuel

The CCC is designated Safety Class for criticality geometry control only.

D4.3.1.1 Core Component Container Safety Function. The CCC is designated Safety Class for criticality geometry control, which is the only safety-class function identified for this SSC. The criticality safety evaluation summarized in Chapter D6.0 demonstrates that criticality is not credible for storage of fuel assemblies within the CCCIISC. The assumption of full water flooding is considered to be not credible and the fixed geometry of the closely packed, CCC, seven-tube arrangement physically restrains the assemblies/pin containers such that criticality is not possible.

The DBAs identified in Chapter D3.0 that could impose structural loads or accelerations on the CCC inside the ISC have the potential to change the criticality geometry control structure (Tables D4-1 and D4-2). Criticality safety evaluation reports (CSERs) discussed in Chapter D6.0 credit the CCC with providing the structure that maintains the critically safe geometry. The functional requirements and evaluations for the CCC (Sections D4.3.1.3 and D4.3.1.4) include summary discussions of the bounding DBAs using the following analyses:

- Handling/drop - Maintain structural integrity within the ISC to provide criticality geometry control after credible drops.

- Cask tipover - Maintain structural integrity to provide criticality geometry control.

- Fuel rod rupture - Maintain structural integrity to provide criticality geometry control up to and including the pressure generated from the rupture of all fuel pins.

- Seismic - Withstand seismic accelerations to maintain structural integrity for criticality geometry control.

- Fire - Withstand transportation design basis fire (DBF) conditions inside the ISC, without exceeding temperature limits for fuel cladding or container components, to maintain structural integrity for criticality geometry control.
D4.3.1.2 Core Component Container Description. The CCC (Figure D2-9) is an unshielded, sealed fuel storage container with seven fuel storage positions. As a defense-in-depth feature, the CCC provides canning for the FFTF fuel during the spent fuel dry storage 40-year design lifetime. The CCC also provides the geometry to ensure criticality control during handling and storage of the fuel. The CCC is designed such that it is fully retrievable from the storage configuration, although the capability to remove individual fuel components from a CCC is not required nor is it guaranteed. The center storage location can accept a fuel assembly that has had the bottom 15.5 in. removed. However, due to the indented CCC grapple handling socket, an Ident-69 pin container cannot be stored in the center location. Therefore, the maximum loading of a CCC will be either six Ident-69 pin containers or seven FFTF fuel assemblies. Pin containers and fuel assemblies have also been analyzed for a mixture of up to five Ident-69 containers and two driver fuel assemblies (DFAs) in a CCC. The Ident-69 container is allowed to contain a maximum of 217 pins, the same number as in a DFA.

The CCC was designed and fabricated in accordance with the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (ASME Code), Section VIII, Division 2 (ASME 1989). The CCC is fabricated from stainless steel and nickel alloy material (approved American Society for Testing and Materials [ASTM] materials for both ASME VIII and ASME III) to provide corrosion-resistant fuel storage, with overall dimensions of 20.0 in. in diameter by 146 in. in height. The weight of the empty CCC is 1,100 lbs. The maximum weight of a loaded CCC occurs with seven DFAs. The weight of the seven DFAs is 3,900 lbs, giving the CCC a maximum gross weight of 5,000 lbs. The storage positions of the CCC are formed by seven steel tubes arranged with six equally spaced, radial storage positions surrounding the seventh center storage position. There is minimal clearance between storage tubes in the bundle orientation. The materials used in the fabrication of the CCC are primarily ASTM-certified 304 stainless steels except for the lower approximately 12-in. portions of the tubes and the lower cups, which are ASTM-certified nickel alloy material UNS06625, and the closure bolts, which are certified to ASTM A-574 (ASTM 1989). The bottom cups provide extra corrosion resistance for each cleaned and dried fuel assembly. These cups are free to move axially (0.12 in.) within the lower support plate to accommodate thermal expansion. The lower support plate provides guidance and spacing for each of the six tube assemblies. The reducing section (saddle) maintains the outside envelope of the container and supports the fuel assemblies.

The upper portion of the outer tubes has a 6.69 in. outside diameter, with 109-mil thick walls. There is a saddle section 14 in. above the bottom of the CCC, where each outer storage tube transitions to a smaller section measuring 4.0 in. in outside diameter, with 226-mil thick walls. The bottom 12.4 in. of the lower section is fabricated from nickel alloy material for enhanced corrosion resistance. The center tube has a 6.54 in. outside diameter, with 120-mil thick walls. The bottom 10.0 in. of the center tube is also fabricated from nickel alloy material. There is no size reduction at the lower end of the center tube.

The outer storage tubes are suspended from the upper support plate. The center storage tube connects the upper support plate with the lower support plate. The outer tubes are fixed only at their upper ends so they are free to accommodate thermal expansion. This design also
permits the outer tubes to stretch slightly to absorb energy during a CCC drop accident onto the ISC internal impact limiter. Drop energy is absorbed by the CCC tubes until the gap between the tube stop and the lower support plate is taken up. The CCC then acts as a rigid body for final interface with the ISC impact limiter during an accident.

The lower support plate has an 18.0-in. diameter and is 1.5 in. thick. It limits downward travel of the outer storage tubes and also provides radial positioning guidance for inserting the CCC into the ISC. The upper support plate has a 20.0-in. diameter, with an overall thickness of 3.6 in. It provides support for the outer storage tubes and the seating surface for the cover seal. There are twelve drilled holes in the upper support plate that accommodate the cover bolts.

The cover has a 20.0-in. diameter and an overall thickness of 1.63 in. The lower surface of the handling socket extends 8.35 in. below the bottom of the cover. Twelve holes are drilled in the cover and are sized to accommodate the closure bolts. The CCC atmosphere is free to pass between the storage tubes, but a metal Helicoflex seal is provided between the container and the cover to establish container confinement. An impact limiter is located on the bottom of the handling socket closure cup. Its function is to reduce the loading to the cover that could occur during the CCC drop accident into the ISC due to the fuel assembly in the center storage location rebounding and hitting the cover. The top surface of the cover is provided with rigging attachment points for empty container handling, a test port for seal leak-rate verification, and a sample port for container atmosphere sampling.

The CCC cover design provides a closure with a cover that mates to the CCC body, crushing a metallic O-ring between them, which provides a primary boundary for storage of the spent fuel. This boundary is not credited for confinement. Leak-testing requirements assure that the CCC will function as an effective "canning" barrier. Each CCC body and closure will be certified at the manufacturer to a leak rate of \( \leq 1 \times 10^{-3} \text{ scc/sec} \). After the CCC is loaded with spent fuel in the interim examination and maintenance (IEM) cell, a pressure decay test to \( \leq 1 \times 10^{-1} \text{ scc/sec} \) is performed on the seal to verify correct installation of the metal Helicoflex seal. Additionally, each fuel storage tube is closed at the bottom with a nickel alloy cup to prevent potential caustic fission product solutions from degrading the ISC confinement liner in the unlikely event of a leak out of a fuel pin. Additional features of the CCC that provide "canning" assurance are as follows: (1) all pressure boundary welds are required to be full penetration and penetrant inspected during the fabrication process; and (2) each CCC is hydrostatically tested to the design pressure of 105 lbs/in² gauge, which is the pressure resulting from 100% fission gas release of seven fuel assemblies.

**D4.3.1.3 Core Component Container Functional Requirements.**

**Handling/Drop** – The CCC is analyzed as an integral part of the FFTF cask/container system. The CCC shall withstand the induced loads from all normal operation and handling/drop accidents. The CCC structural integrity required to maintain criticality geometry control shall not be compromised.
Cask Tipover – The CCC is analyzed as an integral part of the FFTF cask/container system. This analysis is discussed in Section D4.4.1. The CCC structural integrity required to maintain criticality geometry control shall not be compromised.

Fuel Rod Rupture – The CCC is analyzed as an integral part of the FFTF cask/container system. The CCC shall withstand the induced loads from rupture of all pins in seven fuel assemblies. This analysis is discussed in Section D4.4.1. The CCC structural integrity required to maintain criticality geometry control shall not be compromised.

Seismic – The CCC is analyzed as an integral part of the FFTF cask/container system. The CCC shall withstand the 200 Area design basis earthquake (DBE) of 0.26 g, as discussed in Section D4.4.1. The CCC structural integrity required to maintain criticality geometry control shall not be compromised.

Fire – The CCC is analyzed as an integral part of the FFTF cask/container system. The atmosphere internal to the ISC can be either dry argon or dry helium gas. For the different inerting cases evaluated, the maximum inner-cavity wall (liner) temperature limit is 495 °F (argon) or 640 °F (helium), respectively, at the fuel mid-plane (78.5 in. from the top of the CCC) to satisfy a maximum pin cladding temperature limit of 900 °F. The CCC structural integrity required to maintain criticality geometry control shall not be compromised.

D4.3.1.4 Core Component Container System Evaluation. The CCC is designed to "can" the spent fuel. This feature will allow for effective retrieval of the spent fuel in the event of excessive cladding degradation, but does not require a leaktight confinement boundary. This feature was provided for personnel protection during retrieval of the CCC from the ISC after storage. The CCC is not designated as a confinement boundary for storage or on-site shipment. Consequently, a demonstration of confinement requirements was not imposed on the CCC "canning" boundary.

10 CFR 72 requires two independent barriers be maintained between the fuel and the environs. 10 CFR 72.122(h)(1) states that the spent fuel cladding is considered the primary confinement and must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined, "canned," such that degradation of the fuel during storage does not pose operational safety problems with respect to its removal from storage.

As described in a memo to R. A. Almquist (Guttenberg 1995), even though the FFTF spent fuel cladding will meet the requirements noted above upon initial placement into the dry storage cask or ISC, the FFTF spent fuel storage system also "cans" the fuel within the CCC. The function of the CCC is to provide added assurance that the long-term primary boundary of the FFTF spent fuel storage system can maintain performance equivalent to commercial spent fuel cladding. The approach to "can" the fuel within the CCC was implemented due to: (1) the concern that cladding degradation by a phenomenon known as "hot cell rot," which results in fuel
rubblization, may occur during the dry storage life, and (2) a lack of long-term data specific to a
dry cask storage environment for the FFTF spent fuel.

The concern with "hot cell rot" is based on several isolated cases of fuel cladding
degradation that were observed during post-irradiation storage examination of liquid metal
reactor assemblies at several U.S. facilities, as well as in Europe and Japan. In all observed
cases, the phenomenon was strictly associated with degradation of the fuel cladding. The
phenomenon by itself does not produce gross degradation of the oxide fuel. Most documented
instances were associated with storage or handling in air, or where air and moisture in-leakage
was not strictly controlled. "Hot cell rot" was never observed during IEM cell fuel handling at
FFTF.

The cladding degradation mechanism associated with "hot cell rot" is thought to be a
form of caustic stress-corrosion cracking. The irradiated fuel cladding can have large residual
stresses during its storage period, and the presence of minute amounts of residual sodium
hydroxide on pin surfaces and crevices after cleaning with a moist gas-water rinse process is
possible. Exposure to moisture and air could create conditions favorable to stress-corrosion
cracking. The observed cladding degradation associated with "hot cell rot" ranges from scattered
cracks along the fuel pin to cracking around the entire circumference of the fuel pin in the most
severe case. This cracking can result in structural failure during handling of individual fuel pins;
however, the experience to date has been that the cladding provided fuel retention even for the
most severe case.

These isolated failures are not expected to be indicative of how the FFTF fuel will
perform in "leaktight" dry storage. This is because the leaktight ISC limits the in-leakage of
oxygen and moisture impurities to extremely small amounts. Preventing in-leakage of moisture
and air effectively limits the conditions required for stress-corrosion cracking of the cladding to
occur. Extremely small amounts of impurities are contained in the relatively pure inert backfill
gas. These impurities are trapped inside the storage cask when it is sealed and can only react
with a finite number of molecules before the oxidation/degradation reaction stops. Because the
cask is leaktight, it prevents further significant introduction of impurities.

Even though limiting air and moisture in-leakage will effectively limit "hot cell rot"
during dry storage, there remain inherent differences between the FFTF spent fuel and
commercial spent fuel that prevent direct correlation of acceptable dry storage conditions. These
differences include higher burn-ups of FFTF fuel that create different pin pressure characteristics
and more concentrated fission product inventories, different irradiation histories and conditions,
different cladding materials, and the use of sodium versus water coolant. Additionally, there are
no long-term data for storage of FFTF fuel in dry cask storage conditions.

The CCC canning criterion is based on comparison to the expected storage performance
of commercial fuel cladding. Additionally, studies on commercial fuel cladding performance in
long-term dry storage indicate that some failure of fuel rods can be expected in dry storage and
that the resultant defects are in the form of pinholes or hairline cracks that vary in size from
1 to 30 μ (WHC-SD-FF-ES-024). A CCC leak rate of $1 \times 10^{-3}$ scc/sec correlates to a single leakage hole size of 2.5 μ in diameter using American National Standards Institute (ANSI) N14.5, *American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment*, Table B2. This leakage hole size determination is conservative because it assumes that only one leakage path exists (i.e., maximum hole size) and an effective "cladding replacement" boundary is easily demonstrated when compared to the commercially accepted range of 1 to 30 μ for pinhole breaches. Furthermore, the single hole leakage path assumption is extremely conservative and, in reality, the 2.5-μ hole size would be overestimated. In the case of the actual CCC design, any leakage past the seal would be through multiple minor machining marks or imperfections that run perpendicular to the seal surface. Any realistic leakage hole size would be a function of the number of surface imperfections and would be correspondingly smaller.

Further conservatism is demonstrated for particulate retention. The CCC provides a bolted closure seal at the top, and the rest of the assembly is full-penetration welded. The sealed CCC is then leak tested to $1 \times 10^{-3}$ scc/sec. Therefore, it is reasonable to assume that only the seal would have minor leaks based on the unlikely 2.5-μ single leak path hole size discussed above. For a fuel fragment to approach the CCC seal, it would have to overcome gravity and traverse a highly tortuous path up and around the pin plenum region approximately 5 ft in length to the leak path. Additionally, the CCC materials have not experienced the severe environment of a reactor and would not be expected to have degraded properties. The nickel alloy cup closure located at the bottom of each CCC storage tube will retain any potentially damaging caustic mixture and prevent it from contacting the ISC confinement structure in the event of caustic material release outside a fuel pin. The CCC will provide structurally sound retrievability for a case where spent fuel degradation exceeds expectations.

**Handling/Drop** - The CCC was analyzed to confirm that it can safely withstand the following normal, off-normal, and accident conditions. Fully loaded CCCs (six or seven fuel assemblies, or six pin containers) are addressed in WHC-SD-FF-DA-077, *Stress and Structural Analysis of the Core Component Container (CCC)*, and SNF-4790, *200 Area ISA Design Basis Accident Analysis Documentation for FFTF Fuel*. Partially loaded CCCs are not analyzed, and must be evaluated on a case by case basis.

Normal loading conditions:

- Vertical lift and set down by a crane.

Accident conditions:

- 18-ft drop of the CCC into the ISC
- 4-ft sideways drop of the ISC with the CCC inside
- 8-ft drop of the ISC with the CCC inside onto a concrete storage pad.
These conditions are analyzed for the DBA in Chapter D3.0, with the analysis summarized in Section D3.4.2.1. Material properties were evaluated using a peak CCC temperature condition of 600 °F. Normal CCC conditions were evaluated in accordance with the stress limits of the ASME Code, Section VIII, Division 2 (ASME 1989), and the accident conditions were evaluated in accordance with the stress limits of the ASME Code, Section III, Appendix F (ASME 1989).

Nuclear safety equivalency to NRC requirements would require the CCC to be designed and fabricated to ASME Code, Section III, Subsection NB or NC, for providing criticality geometry control. This was identified during final reviews for initial loading of the FFTF fuel in 1996. A decision was made, with DOE approval, to continue with the ASME Section VIII container since it provided sufficient structural capacity for storage purposes. It was acknowledged that the FFTF fuel would require repackaging into an approved ASME Section III container prior to off-site transport for final disposal at a repository. This repackaging was assumed to occur in a future "hot cell" to be co-located near the 200 Area ISA and Canister Storage Building facilities.

A thermal stress analysis was performed on the CCC. The results indicate that the average temperature around the circumference of the storage tube at the mid-plane of the core component fuel section is well below the 600 °F design temperature used to bound the structural calculations. The thermal analysis, WHC-SD-FF-ER-100, Thermal Analysis of the Core Component Container Within the Interim Storage Cask, shows that both normal and accident conditions will result in ISC liner temperatures that are well below the limiting design requirement parameter of 495 °F. The CCC structural integrity required to maintain criticality geometry control was not compromised.

**Cask Tipover** – The CCC is an integral part of the FFTF cask/container system and is analyzed for the DBA in Chapter D3.0. This analysis is summarized in Section D3.4.2.3 and discussed in Section D4.4.1. Analyses performed for the ISC tipover show that induced stresses and loads are bounded by the 4-ft drop onto an unyielding surface analyzed in Section D3.4.2.1. The CCC structural integrity required to maintain criticality geometry control is not compromised.

**Fuel Rod Rupture** – The accident condition of all fuel rods being ruptured is analyzed for the DBA in Chapter D3.0, and the analysis is summarized in Section D3.4.2.4. The inventory of fission gas and those gases produced during 50 years of radioactive decay are analyzed. The analysis assumes the worst-case ISC cavity average temperature of 286 °F. The maximum pressure is determined to be 62 lbs/in² gauge for the CCC worst-case loading of seven DFAs. The design pressure of the CCC is 70 lbs/in² gauge, thus providing a pressure margin of 8 lbs/in² or about 11%. A finite element stress analysis was also performed for the pressure load of 70 lbs/in² gauge (WHC-SD-FF-DA-077). The analysis determined that CCC structural integrity required to maintain criticality geometry control is not compromised. Each CCC is hydrostatically tested to the design pressure of 105 lbs/in² gauge.

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Annex D – 200 Area Interim Storage Area
Seismic — The design basis seismic event for the CCC, as an integral part of the FFTF cask/container system, is analyzed for the DBA in Chapter D3.0 and the analysis is summarized in Section D3.4.2.5. The CCC structural integrity required to maintain criticality geometry control was not compromised as the ISC does not slide or tip over. All seismically induced loads are well bounded by the drop impact accelerations.

Fire — The conditions of peak thermal temperature during and after the transportation DBF are analyzed for the DBA in Chapter D3.0. These analyses are summarized in Section D3.4.2.7. The fuel cladding temperature limit of 900 °F and CCC closure seal temperature limit of 500°F are evaluated.

The accident fire results in a short-term thermal loading on the ISC. Section D3.4.2.7 summarizes the ISC temperatures in the cask during a fire. Most of the concrete remains below the 350 °F accident temperature allowable by American Concrete Institute (ACI) 349-90, Code Requirements for Nuclear Safety Related Concrete Structures. For the conditions representing the fire accident, the large mass of concrete helps keep the interior temperatures of the cask relatively cool. The maximum liner temperature reaches 309 °F, within the allowable limit of 495 °F (with argon-inerting gas). At least 17 in. of concrete remain below the 350 °F ACI 349-90 temperature limit defined for accident conditions.

D4.3.1.5 Core Component Container Controls (Technical Safety Requirements). The assumptions associated with the CCC that require TSRs to ensure performance of its safety function are as follows:

- Each CCC shall contain six DFAs in the outer positions, seven DFAs including the center position, or six Ident-69 containers in the outer positions. The loaded CCC can contain a mixture of fuel assemblies and pin containers such that the total is either six or seven, and the number of pin containers is five or less. No partially filled CCCs are permitted.
- There shall be no more than a total of 1,519 pins in a CCC container to protect the fuel rod rupture pressure calculation assumptions.
- Only intact fuel shall be placed into a CCC.

D4.3.2 Light Water Reactor Canister

The LWR canister structure provides required criticality geometry control and is therefore designated Safety Class, as identified in Table D4-2. The canister also has safety-significant functions, as discussed in Section D4.4.5.

The Nuclear Assurance Corporation (NAC)-1 inner LWR canister design performance requirements are established by SNF-4894, Spent Nuclear Fuel Project Acceptance Criteria for
LWR Fuel Storage System. The following discussions are based on a canister design that is in the final stages of design, review, and approval. The design provides a welded closure and dimensional constraints to accommodate future packaging in standard containers. Characteristics of the canister (e.g., design pressures or structural strength) are expected to be unchanged from or improve upon the canister design used as the basis for evaluation. Calculations and text in this section will need to be verified upon design completion and approval of a design analysis report for the LWR canister. Revision will be required only if the characteristics and features of the canister are changed significantly. All affected discussions will be reviewed for applicability based on the final LWR canister design.

D4.3.2.1 Light Water Reactor Canister Safety Function. The LWR canister provides the structure that ensures criticality geometry control of the LWR fuel during storage at the 200 Area ISA and is designated Safety Class for criticality prevention only. No other safety-class functions were identified by the accident analysis (Chapter D3.0) for the NAC-1 cask system components. The analyses did identify safety-significant functions of the inner canister. The safety-significant functions for individual components of the NAC-1 cask system are discussed as an integral part of the NAC-1 cask evaluation provided in Section D4.4.5. This includes the inner canister, the NAC-1 cask, and the International Standards Organization (ISO) shipping/storage container. Specific individual component evaluation is addressed in Section D4.4.5.4.

The DBAs identified in Chapter D3.0 that could impose structural loads or accelerations on the LWR canister inside the NAC-1 cask have the potential to change the criticality geometry control structure (Tables D4-1 and D4-2). CSERs discussed in Chapter D6.0 credit the inner canister with providing the structure that maintains the critically safe geometry. The functional requirements and evaluations for the LWR canister (Sections D4.3.2.3 and D4.3.2.4) include summary discussions of the bounding DBAs using the following analyses:

- Handling/drop – Maintain confinement of radioactive materials after a credible drop (Safety Significant) within the NAC-1 cask and maintain structural integrity to provide criticality geometry control (Safety Class).
- Fuel rod rupture – Maintain confinement of radioactive materials (Safety Significant) and maintain structural integrity to provide criticality geometry control (Safety Class) at the pressure generated.
- Seismic – Withstand seismic accelerations without loss of confinement (Safety Significant) and maintain structural integrity to provide criticality geometry control (Safety Class).
- Fire – Withstand transportation DBF conditions inside the NAC-1 cask without loss of confinement (Safety Significant) or exceeding temperature limits for fuel cladding.
or container components to maintain structural integrity for criticality geometry control (Safety Class).

The NAC-I cask provides a passive structure that ensures the integrity of the inner canister structure will be maintained.

**D4.3.2.2 Light Water Reactor Canister System Description.** The commercial LWR fuel storage system consists of six storage units, each comprised of an LWR canister, a NAC-1 cask, and an ISO shipping/storage container. The welded inner canister and the intact fuel cladding will provide confinement of the fuel during storage, as required by 10 CFR 72. The NAC-1 cask provides a shielding overpack that provides weather and NPH protection.

The LWR canisters are designed and fabricated to the requirements of the ASME Code Section III, Subsection NB (ASME 1995), as required by the LWR canister acceptance criteria. Canister internals, provided for positioning the fuel assemblies, and the container for consolidation of the loose rods fall under ASME Code, Section III, Subsection NG. The canisters were designed for a life expectancy of 50 years.

The LWR canister provides a confinement boundary for the 300 Area LWR fuel. The canister is fabricated from a 12-in. stainless steel pipe with welded base cap and top closures. It has a nominal outside diameter of 12.75 in., with a maximum outside diameter of 13.0 inches at the end cap, and a total length of 165.25 in. The bottom cap is machined, so the pipe-to-cap weld is inspectable at the side of the canister. The closure lid contains a penetration that allows the canister to be evacuated, filled with helium, and leak tested to the requirements of ANSI N14.5.

The canisters provide criticality geometry control such that a loaded canister will have a $k_{eff}<0.95$ when fully moderated and reflected. A 2.0-in. thick stainless steel removable handling fixture is provided for the canister upper cap. Space remains in the cask cavity for an additional 10 in. of gamma shielding.

The design weight of the canister is 1,250 lbs. The maximum loaded weight of the canister is not to exceed 3,300 lbs. The maximum internal design pressure of the canister is 75 lbs/in² gauge testable to a pressure of 100 lbs/in² gauge.

The canister used for loose boiling water reactor (BWR) and pressurized water reactor (PWR) rods incorporates an internal spacer arrangement to contain the rods (Figure D2-14). The fuel is loaded in a split 8-in. stainless steel schedule 80 pipe that is skip-welded closed. This container extends into the canister lid recess for upper external restraint and is packaged in the canister cavity within a sleeve of 10-in. stainless steel schedule 40 pipe.

The NAC-1 cask provides shielding and protection of the inner canister. The structures of the NAC-1 cask are constructed of stainless steel. The cask cavity is formed by the inner shell, which is a 14.125-in. outside diameter, 0.3125-in. thick stainless steel shell. The upper end of the shell is welded to the cask cavity flange; the bottom end of the shell is welded to the cask
bottom casting. Surrounding the inner shell of the cask is a nominal 6.6875-in. thick annulus of chemical-grade lead (gamma) shielding. The lead is shaped such that approximately 5 in. from the bottom and 30 in. from the top, the thickness is reduced to 5.4375 in. There is an annular void, 5 in. long by 1.25 in. thick, at the bottom end of the gamma shield to allow for any lead expansion during the fire accident. The upper axial shaping is accomplished by reducing the diameter of the outer shell 2.50 in. over a 30-in. length. The lead/steel interface of the inner and outer shell have axial copper fins that are imbedded in the lead and welded to the inner and outer shells to transfer heat across the interface with a minimum temperature gradient (see Figure D2-10). Additional NAC-1 cask design and fabrication details are provided in Section D4.4.5.2.

The 300 Area LWR irradiated fuel inventory addressed in this document consists of five PWR assemblies and consolidated BWR and PWR individual rods. Each PWR assembly is packaged in an individual cask in the assembly configuration. The individual rods are packaged in a single cask in a container that fits within the inner canister. Characterization of the fuel inventory is discussed in Section D2.5.1.3.

D4.3.2.3 Light Water Reactor Canister Functional Requirements

Handling/Drop – The inner canister is analyzed as an integral part of the NAC-1 cask/container system. The NAC-1 cask/container system, including the canister, shall withstand the induced loads from all normal operation, handling, and storage drop accidents sufficient to ensure that the inner canister structural integrity required to maintain criticality geometry control shall not be compromised.

Fuel Rod Rupture – The canister shall withstand the internal pressure resulting from rupture of all contained fuel rods without compromising the canister structural integrity required to maintain criticality geometry control.

Seismic – The inner canister is analyzed as an integral part of the NAC-1 cask/container system. The NAC-1 cask/container system, including the canister, shall withstand any induced stresses resulting from the design basis seismic event, with sufficient integrity to ensure that the inner canister structural integrity required to maintain criticality geometry control is not compromised.

Tornado/wind – Withstand tornado pressure differential without structural damage to the LWR canister, which could result in loss of criticality geometry control.

Fire – The inner canister is analyzed as an integral part of the NAC-1 cask/container system. The NAC-1 cask/container system, including the inner canister, shall withstand temperatures resulting from the DBF without compromising the canister structural integrity required to maintain criticality geometry control. This includes a requirement to not exceed the maximum allowable fuel cladding temperature limit to assure that the fuel remains in the analyzed geometry.
D4.3.2.4 Light Water Reactor Canister Evaluation. The LWR canister provides the structure that ensures critical geometry control of the LWR fuel during storage at the 200 Area ISA and is designated Safety Class for criticality safety reasons only. The criterion for criticality safety is that $k_{\text{eff}} < 0.95$ be demonstrated for all conditions, incorporating any bias/uncertainties associated with the analysis.

Two CSERs were prepared for the interim storage of the NAC-I casks in the 400 Area ISA. Chapter D6.0 addresses specific criticality aspects of 200 Area LWR fuel storage at the ISA as individual casks and in the storage array. The LWR canister is required for criticality geometry control and is therefore designated Safety Class. Structural integrity of the PWR fuel assembly is required by the criticality analysis to retain the fuel material within the square cross-sectional geometry of the fuel assembly contained in the inner canister. Specific structural analysis of PWR assemblies need not be performed since this has been addressed for commercial fuels in a Lawrence Livermore National Laboratory document UCID-21246, Dynamic Impact Effects on Spent Fuel Assemblies. The document was prepared to assess the effects of dynamic impacts (to be expected from cask drop or similar incidents) on the integrity of fuel rod cladding for zircaloy-clad LWR spent fuel assemblies during cask handling and storage. The study concluded that commercial LWR fuel assemblies will not yield or buckle at the impact g loads calculated for the NAC-1 cask, and their structural integrity can be assumed for safety purposes. The canisters for the PWR assemblies, however, provide defense in depth for criticality safety due to the provision of safety-class support for retention of the fuel assembly geometry.

The consolidated individual BWR and PWR fuel rods that are stored in a NAC-1 cask are packaged in a canister with internal components that retain the cross-sectional geometry limits required for criticality control. The loose rods are packaged in a split container fabricated from 8-in. schedule 80 stainless steel pipe. The split container is tack welded shut and packaged in the canister within a 10-in schedule 40 stainless steel pipe sleeve. The 10-in. pipe provides the limiting diameter for criticality geometry control. The individual rods remain oriented in a close packed restricted configuration similar to the rods in the fuel assemblies and are bounded by the assembly buckling loads and conclusions.

Handling/Drop – The LWR canister is analyzed as an integral part of the cask/container system in which the NAC-1 cask provides a passive barrier to stresses resulting from storage/handling drops. Evaluation of the cask/container system handling/storage drop-induced stresses is provided in Section D4.4.5. The NAC-1 SAR (NAC 1990) provides evaluations that examine hypothetical accidents that conservatively bound the 200 Area ISA accident-induced stresses. The details of the analysis are provided in Chapter D3.0 and summarized in Section D3.4.2.1. As required by 10 CFR 71, "Packaging and Transportation of Radioactive Material" (1995 edition), calculations were documented for the types of analyses discussed below. The original NAC-1 SAR-referenced sections of 10 CFR 71 (1971 edition) are no longer directly citable; however, the requirements are the same as those listed in the following statements.
Cask analyses evaluated to hypothetical accident conditions (10 CFR 71.73) include the following:

- Analyze a free drop of the cask through a distance of 30 ft onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected;

- Followed by a free drop of the cask through a distance of 40 in., in a position for which maximum damage is expected, onto the upper end of a solid, vertical, cylindrical, mild steel bar mounted on an essentially unyielding, horizontal surface. The bar must be: (1) 6 in. in diameter, with the top horizontal and its edge rounded to a radius of not more than 0.25 in., and (2) of a length as to cause maximum damage to the cask, but not less than 8 in. long. The long axis of the bar must be vertical.

The design of the NAC-1 cask system is predicated on the ability of the cask to transport 150-day cooled PWR spent fuel. The thermal energy of the design basis PWR fuel imposed temperature and pressure conditions on the cask structure that produced stresses that were significantly greater than the stresses imposed by the thermal energy and pressure of the 300 Area LWR fuel rods in the inner container. The calculations provided for the NAC-1 transportation calculations are therefore conservative for 200 Area ISA storage conditions.

Evaluation of the 400 Area ISA storage configuration included results from the NAC-1 SAR (NAC 1990) and additional analyses specific to the ISA. The evaluation of the analyses summarized in Section D.3.4.2.1 demonstrated the following:

- Conclusion for the 30-ft free fall analyses – The NAC-1 SAR analyses concluded that the integrity of the NAC-1 cask would be maintained, although significant damage would occur to the exterior, sacrificial structures of the NAC-1 cask.

- Conclusion for the 40-in. free fall onto a mild steel pin analyses – The NAC-1 SAR analyses concluded that the integrity of the NAC-1 cask would be maintained. Minimal damage to the cask, namely the localized deformation and/or slight puncture of the shield tank, would be incurred.

The LWR canister was designed and analyzed in accordance with the requirements of the ASME Code, Section III, Subsection NB (ASME 1995), as required by NUREG/CR-3854, Fabrication Criteria for Shipping Containers, for the Category I component safety group and 10 CFR 71. The canisters are designed for an operational environment inside the NAC-1 cask body under both normal and accident conditions of transport per 10 CFR 71, and normal and off-normal conditions of storage per 10 CFR 72.

The analysis of the LWR canister for the postulated accident conditions of transport investigated the integrity of the canister to a 96 g axial impact deceleration load and the probable initiation of buckling. Fatigue analysis was also performed to predict inner canister limitations to
mechanical and thermal cyclic loading conditions. The analysis shows that the design of the LWR canister satisfies the required design checks with an acceptable margin of safety to ensure that the stresses induced will not compromise the canister structural integrity required to maintain criticality geometry control.

**Fuel Rod Rupture** – The fuel-rod rupture analysis is provided in Chapter D3.0, and the results are summarized in Section D3.4.2.4. The Chapter D3.0 analysis includes calculation of the maximum normal operating pressure for the LWR canister and assumes cask cavity temperature as a result of the hottest summer day (115 °F ambient), along with fuel rod rupture of all rods. The pressure increase within the cask cavity is due to both a temperature increase and 100% rod rupture. This pressure is calculated in WHC-SD-FF-DA-087, *NAC-1 Cask/Canister Pressure Evaluation for PWR Fuel, 300 Area LWR Fuel Storage*, which addresses the fission gas generated during irradiation, initial pressurization within the rods, and the gas within the canister cavity from loading and leak-testing activities.

The pressure within the cavity due to rod rupture at maximum normal conditions is within the design pressure of 75 lbs/in² gauge and pneumatic test pressure of 100 lbs/in² gauge established for the canister.

Worst-case pressures are also calculated for 100% rod rupture during the DBA fire conditions. This accident condition pressure is also well within the design pressures for both the inner canister and the NAC-1 cask, and the stresses induced will not compromise the canister structural integrity required to maintain criticality geometry control.

**Seismic** – The seismic analysis is provided in Chapter D3.0 and is summarized in Section D3.4.2.5. As noted, stresses induced in the NAC-1 cask and the LWR canister during the 200 Area ISA DBE are within the cask system design capabilities. The LWR canister safety-class evaluation calculations consider all components to be part of an integral NAC-1 storage system.

Seismically induced loads are bounded by the drop impact accelerations above. The drop impact evaluation concludes that the cask system is able to withstand any induced stresses from handling/drop events without compromising the structural integrity of the LWR canister required to maintain criticality geometry control.

**Tornado** – The LWR canister design pressure is much greater than the 0.9 lb/in² gauge pressure differential of the design basis tornado (DBT). This minimal differential pressure cannot generate structural damage to the LWR canister.

**Fire** – An analysis of the thermal response of the storage system to the DBF is provided in Chapter D3.0 and summarized in Section D3.4.2.7. The NAC-1 cask SAR (NAC 1990) determined that the cask supported an internal thermal load in excess of the LWR fuel and presented analysis of the transportation fire event that concluded that the cask integrity is preserved. That analysis provides a bounding case for use of the NAC-1 cask for storage of
200 Area LWR fuel. The Chapter D3.0 analysis concludes that the stresses induced will not compromise the canister structural integrity required to maintain criticality geometry control.

**D4.3.2.5 Light Water Reactor Canister Controls (Technical Safety Requirements).** The assumptions associated with the LWR canister that require TSRs to ensure performance of its safety function of criticality geometry control are as follows:

- The number of individual fuel rods in a LWR canister does not exceed a maximum of 179 PWR rods, or 96.5 BWR rods consolidated with 17 PWR rods.
- The individual loose rods must be retained within the 10-in. diameter sleeve (design feature).
- Only intact fuel shall be placed into an LWR canister.

**D4.4 SAFETY-SIGNIFICANT STRUCTURES, SYSTEMS AND COMPONENTS**

**D4.4.1 Fast Flux Test Facility Interim Storage Cask System**

This section discusses the safety significant aspects of the ISC and CCC. The criticality safety aspects of the CCC, which relate to safety-class considerations, are discussed in Section D4.3.1.

**D4.4.1.1 Safety Function.** The ISC is designed to provide secondary confinement for the fuel and structural protection for the CCC. It also provides passive heat removal and radiological shielding protection for site personnel by limiting the dose rate at normally accessible surfaces to acceptable levels (not credited safety functions). As shown in Table D4-2, the ISC is credited with safety-significant functions for the following DBAs identified in Chapter D3.0:

- Handling/drop – Maintain confinement of radioactive materials after a credible drop and provide passive protection of the CCC such that it retains structural integrity.
- Mobile crane fall – Maintain structural integrity sufficient to maintain confinement of radioactive materials after a crane fall.
- Cask tipover – Maintain confinement of radioactive materials after the crane fall and provide passive protection of the CCC such that it retains structural integrity.
- Fuel rod rupture – Maintain confinement of radioactive materials after the rupture of all fuel pins.
Seismic – Withstand seismic accelerations without loss of structural integrity sufficient to maintain confinement, and without tip over or sliding.

Tornado/wind – Withstand DBT winds (excluding DBT missiles) without sliding or tip over. Also withstand the design basis wind and wind-driven missiles established for the 200 Area ISA without loss of structural integrity sufficient to cause loss of confinement.

Fire – Withstand transportation DBF conditions without loss of structural integrity sufficient to cause loss of confinement or exceeding temperature limits for fuel cladding or cask components.

In addition, the ISC provides a confinement boundary for all normal conditions and abnormal events.

As shown in Table D4-2, the CCC is credited with safety-class criticality control functions for the DBAs identified in Chapter D3.0 and previously discussed in Section D4.3.1, as follows:

- Handling/drop
- Cask tipover
- Fuel rod rupture
- Seismic
- Fire

D4.4.1.2 System Description. The ISC is an aboveground concrete and steel shielded, top-loading spent fuel storage cask that will be used to provide safe interim dry storage of a CCC with FFTF spent fuel assemblies or pin containers for a period of up to 40 years at the 200 Area ISA. One CCC can be stored in the cavity of each ISC. The ISC has been designed and fabricated to meet the requirements of WHC-S-4110, Specification for FFTF Interim Storage Cask, in accordance with 10 CFR 72. "Canning" of the spent fuel is provided by the CCC, as discussed in Section D4.3.1. The ISC is designed to provide secondary confinement for the fuel and environmental protection for the CCC. The ISC also provides passive heat removal and radiological shielding protection for site personnel by limiting the dose rate at normally accessible surfaces to acceptable levels. A weather protection cover is installed on each ISC in the ISA.

The ISC design consists of an all-stainless steel internal confinement structure surrounded by steel and concrete shielding. The fully loaded cask, including a loaded CCC with a gross payload of 5,000 lbs, the closure hardware and the weather cover, weighs a maximum of 114,200 lbs. Outer cask dimensions are 85 in. in diameter by 181 in. tall, excluding the weather cover. The internal cavity of the ISC is 21 in. in diameter and 147 in. tall, and will accept one CCC. This cavity, which is formed by a 1.5-in. thick stainless steel cylinder and 8-in. thick bolted top and welded bottom closure plates, provides the confinement boundary. An additional
cover plate may be seal welded over the bolted closure after receipt at the 200 Area ISA to enhance the long-term storage configuration. This cover plate would be seal welded to the vertical cylindrical stainless steel liner around and above the bolted shield closure plug. The lower end of this existing liner is welded to the top confinement flange of the ISC cavity, and there is a 3-in. recessed space above the bolted closure below the existing weather cover. Although it is not anticipated that this cover plate would be removed during the storage period, the seal weld located in this space is accessible with the weather cover removed and could be readily ground out to restore the ISC to its original configuration for future repackaging or for access to test ports in the closure plug.

The ISC confinement boundary design and analysis were performed by General Atomics, which holds an ASME Certificate of Authorization N for design and overall fabrication responsibility for ASME Code, Section III, Division 1 and 2 components. The ISC design analysis and material properties were based on the requirements of the ASME Code, Section III, Subsection NC (ASME 1989). Fabrication of the ISC is based on the ASME Code, Section III, Article NC-4000. The ISC confinement boundary is constructed of ASTM materials but was not required to be stamped or built by a certificate holder. Examples of using the ASME Code, but not requiring a code stamp, can be found in NUREG/CR-3854 and NUREG/CR-3019. Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials. The fabricator was not required to be a certificate holder as the design was performed by a vendor with an ASME Certificate of Authorization N for design and overall fabrication responsibility for ASME Section III, Division 1 and 2 components, and the N stamp was not required. Additionally, all ISC design and fabrication activities were performed using a 10 CFR 72 quality assurance program or equivalent.

The bottom of the ISC cavity is fitted with an aluminum crush pad to limit CCC impact loads in the unlikely event that it is dropped into the ISC during loading. The surfaces of the cavity are finished to remove burrs, sharp corners, and weld beads that could potentially interfere with cask loading operations. The confinement structure described above is also surrounded by annular steel shield plates that are surrounded by concrete reinforced with rebar. As discussed in the FFTF ISC design analysis report (General Atomics 1995), the concrete shielding structure is designed to meet ACI 349-90 requirements. The concrete design mix was selected to ensure strength and long life over the range of temperature conditions expected during normal operations, and the more extreme short-term temperatures that could occur during off-normal or accident conditions.

The ISC heat dissipation is totally passive. All required heat removal can occur by conduction, thermal radiation, and convective cooling of the outer surface. The ISC is also provided with a passive cooling system that removes heat by using an internal natural circulation airflow system. The airflow system is formed by two inlet ducts, an annular gap between the confinement boundary cylinder and the inner shield, and two outlet ducts. The inlet and outlet ducts are steel-lined penetrations through the concrete that take non-planar paths to minimize radiation streaming. These penetrations consist of two 4-in. outside diameter ducts that supply air to the bottom of a 0.75-in. wide annulus between the confinement boundary cylinder and the
inner shielding cylinder. Natural convection circulates air up the annular space and out two similar ducts at the top of the annulus. This overall heat removal system is designed to limit localized concrete temperatures to below 200 °F on the warmest normal condition day, with the bulk of the concrete maintained below 150 °F. The ISC thermal analysis for the extreme ashfall accident (General Atomics 1995) shows that even if the ducts become plugged and natural convection cooling is eliminated, these limits will still be met. Therefore, operability of the cask ventilation system is not required to be monitored to ensure that cask thermal limits are not exceeded.

Based on the white-painted exterior surface of the ISC, the solar absorptivity of the cask is taken to be 0.3 and the emissivity is 0.9. The emissivity value is based on heavily oxidized paint. The only possible effect that could result from degradation of the paint is a possible increase in the absorptivity, but this case is bounded by the analyzed ashfall accident case (General Atomics 1995), which demonstrates normal limits are not exceeded. Therefore, the ISC paint color is not a safety requirement. The off-normal severe cold is used to evaluate the thermal distributions of the ISC for the -27 °F lowest ambient temperature reported for the Hanford Site. All temperatures are within limits and the thermal gradients are less than the normal condition case with no wind.

The ISC provides several layers of shielding to ensure worker as-low-as-reasonably achievable (ALARA) radiation protection. The first layer of shielding consists of a 3.0-in. thick carbon steel clamshell around the entire length of the cavity. Another partial length, 4.0-in. thick, carbon steel clamshell shield is provided for additional shielding at the cavity mid-plane where the fuel section is located. The clamshell shields are designed with studs that attach to the reinforced concrete cylinder shield. This shield is a minimum of 21.25 in. thick for additional radial shielding. Supplemental axial shielding is provided by 4.0-in. thick plates below the bottom head and below the upper closure. A design dose rate of 2 mrem/h at accessible surfaces for normal conditions and 200 mrem/h at the (inaccessible) bottom head is defined in WHC-S-4110. Shielding acceptance criteria in the design specification allowed a localized dose rate of 5.0 mrem/h to account for potential shielding imperfections and localized hot spots.

The design description of the CCC is presented in detail in Section D4.3.1.

D4.4.1.3 Fast Flux Test Facility Interim Storage Cask System Functional Requirements. The ISC containing a CCC shall be able to withstand the accident loads associated with handling and drops, mobile crane falls, tipovers, fuel rod ruptures, seismic events, tornado/wind events, and fire/thermal events. In addition to those loads, additional handling accident loads were evaluated to fully envelop all ISC load conditions. The ISC shall withstand these design-basis accident loads to the extent that the reduction in shielding is not sufficient to increase the external dose rate to more than 1,000 mrem/h at 1 m from the external surface of the package specified in WHC-SP-4110, and the ISC confinement integrity is maintained (i.e., stress levels in the ISC confinement boundary do not exceed the levels specified for Level D service limits of the ASME Code, Section III [ASME 1989]). Each type of accident load is discussed below.
Confinement – Two confinement boundaries shall be maintained for the fuel in normal and accident conditions. The accident loads and conditions under which confinement must be maintained are identified in Section D4.4.1.1. The ISC shall be one of the confinement boundaries and the intact fuel cladding shall be the other confinement boundary. The cask must be designed to provide redundant sealing of confinement systems.

Handling/Drop – The FFTF cask system shall be analyzed to remain functional with no loss of confinement or structural integrity for the following handling and drop scenarios:

- A free drop of 40 in. striking the top end of a vertical, cylindrical, mild steel bar mounted on an unyielding horizontal surface in the worst-case orientation.

- A free drop of 4 ft onto a flat, unyielding, concrete horizontal surface, striking the surface in a position for which maximum damage is expected.

- A free drop of 8 ft onto a flat, 1.5-ft. thick reinforced concrete horizontal surface, striking the surface in a position for which maximum damage is expected.

- A drop of a fully loaded CCC from a height of 18 ft into an ISC.

Loads shall be combined such that normal loads from pressure, temperature, and dead weight act in combination with all other loads. No two off-normal or accident events are postulated to occur simultaneously.

Mobile Crane Fall – The ISC shall be able to withstand direct impact of the mobile crane boom without loss of structural integrity sufficient to maintain confinement if a crane failure occurs.

Cask Tipover – The ISC shall be able to withstand the induced load from a tipover accident without loss of confinement, loss of shielding beyond accident limits, or loss of criticality configuration geometry control by the CCC.

Fuel Rod Rupture – The ISC design pressure of 36.7 lbs/in² gauge shall not be exceeded by fuel rod rupture regardless of thermal conditions.

Seismic – The ISC containing the CCC shall be able to withstand the 200 Area ISA DBE without loss of structural integrity sufficient to maintain confinement and without sliding or tip over.

Tornado/Wind – The ISC shall be able to withstand the DBT winds (excluding missiles) without loss of structural integrity sufficient to maintain confinement and without sliding or tip over. The ISC shall also be able to withstand the design basis wind and wind-driven missiles established for the 200 Area ISA.
Fire - The design basis storage fire is bounded by the design basis transportation fire of 10CFR71.73(c)(3) per the 200 Area ISA FHA (HNF-4932), with the following requirements:

- Fuel cladding temperatures shall not exceed 900 °F to preclude cladding breach.
- The cask shall be designed to provide adequate heat removal capacity without active cooling systems (10 CFR 72.236[f]).
- The atmosphere internal to the ISC can be either dry argon or dry helium gas. For each inerting case that is evaluated, the inner-cavity wall (liner) maximum temperature limit is 495 °F (argon) or 640 °F (helium) at the fuel mid-plane (78.5 in. from the top of the CCC) to satisfy the maximum pin cladding temperature limit of 900 °F. The acceptance criteria for the ISC peak liner temperature was based on pre-design analyses that were performed to allow the ISC design to proceed independent of a CCC thermal analysis. These criteria were established (WHC-SD-FF-ES-024) based on the 900 °F cladding limits. The ISC liner limits were determined by analysis of various inerting cases based on a worst-case concrete and steel shielded, non-ventilated, storage cask configuration containing a CCC loaded with six 250-W assemblies. For this configuration, ISC liner limits of 495 °F with argon-inerting gas, or 640 °F with helium-inerting gas, were chosen to ensure that the 900 °F cladding temperature limit would not be exceeded for any cask design. Since the CCC is inerted with argon, the 495 °F limit is conservatively imposed for the ISC liner.
- The concrete temperature limits for the ISC consist of normal localized, average, and accident temperature limits in accordance with guidelines of ACI 349-90. The ACI long-term concrete normal temperature limits are 150 °F for average temperatures and 200 °F for localized areas. The short-term accident temperature limit is 350 °F.
- The Helicoflex seal temperature limits are 700 °F for the ISC and 500 °F for the CCC. These limits are established based on the manufacturer's recommendation for the different seal materials.
- Complete blockage of the ISC ventilation ducts shall not result in temperatures that exceed any functional requirements listed above.

D4.4.1.4 Fast Flux Test Facility Interim Storage Cask Evaluation.

Confinement - The FFTF spent fuel storage system is provided with multiple systems to confine the radioactive fuel. The ISC serves as the main pressure boundary and leakage barrier to the environment to ensure that environmental, worker, and public safety is maintained and to control the storage atmosphere such that degradation of the spent fuel remains limited. The maximum permissible design leakage rate for the ISC confinement barrier is "leaktight," as defined in ANSI N14.5. At this leak rate, $1 \times 10^7$ scc/sec air, the stored spent fuel is not expected to degrade during long-term dry storage, as the leaktight boundary prevents oxygen and
moisture from entering the cask and contaminating the inert atmosphere. As such, there is no known mechanism for accelerated corrosion of the spent fuel. The FFTF spent fuel cladding must meet the intact fuel criteria when the fuel is placed into the CCC. Because each spent fuel assembly will be washed using the sodium removal process in the IEM cell, an assessment of cladding integrity is achieved before transfer to dry storage. The sodium removal wash water will provide indication of a gross cladding failure.

There are no long-term data specific to dry storage of FFTF spent fuel in the leaktight inert dry atmosphere conditions of a dry storage cask. The ISC design is consistent with commercial spent fuel dry storage practice and restricts oxygen and moisture in-leakage to minute levels such that degradation of the spent fuel is expected to be effectively controlled over the storage lifetime. Even so, because there is no long-term data, it is assumed that the FFTF spent fuel cladding may not perform in the same manner as commercial fuel. This results in the requirement to "can" the spent fuel assemblies within the CCC.

The CCC boundary consists of a cluster of seven closed-bottom tubes that form the storage basket, with a bolted closure that contains a single metal seal. The CCC body and closure seal assembly are tested and certified at the manufacturer to provide a $<1 \times 10^{-3}$ scc/sec leakage rate, in accordance with ANSI N14.5. Additionally, the final closure assembly seal, located at the top of the CCC approximately 5 ft above the fuel zone, is tested to a leak rate of $<1 \times 10^{-3}$ scc/sec after loading the spent fuel in the IEM cell. The assembly test is in addition to visual verification that the seal is correctly installed. This approach ensures the fuel is effectively " canned" and exceeds the gross visible rupture criteria for cladding confinement that is allowed for commercial spent fuel.

10 CFR 72.236(e) requires the ISC to be designed for redundant sealing of the confinement systems. The ISC is designed with redundant seals in the closure lid and redundant welds on the penetration port cover (10 CFR 72). Confinement system redundancy for the ISC is ensured by a combination of inspection techniques, which include radiographic and ultrasonic inspection, helium leak testing, and dye penetrant testing of the confinement welds. The confinement capability of the empty ISC liner assembly is assured by the manufacturer by radiographic inspection of the longitudinal and circumferential full penetration welds, and by ultrasonic inspection of the liner to upper flange and bottom plate full-penetration welds. In addition to these tests at the manufacturer, a complete helium leak test of the entire confinement liner is performed. The confinement capability of the loaded ISC is assured by helium leak-testing both closure seals after assembly and dye penetrant testing both seal welds of the penetration port cover plates.

Tests and specifications relevant to confinement integrity of the ISC at the time of cask loading include the following:

- ISC helium backfill pressure requirements
- ISC dye penetrant test of the penetration port closure welds
- ISC maximum permissible leak rate.
10 CFR 72.236(j) required the ISC to be inspected to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce its confinement effectiveness. The quality standards under which the ISC was fabricated and welded provide the assurance of confinement integrity. Additionally, the ISC was pressurized and leak tested after all confinement welding was completed.

A thermal analysis states the atmosphere internal to the ISC can be either dry argon or dry helium gas. For the different inerting cases evaluated, the maximum inner-cavity wall (liner) temperature limit is 495 °F (argon) or 640 °F (helium), respectively, at the fuel mid-plane (78.5 in. from the top of the CCC) to satisfy a maximum pin cladding temperature limit of 900 °F to preclude cladding breach. The thermal analysis shows that the bounding case is reached during the ashfall case in which the peak cladding temperature reaches 800 °F (WHC-SD-FF-ER-100). The maximum seal temperature for the CCC reaches 196 °F during the ashfall condition, as indicated in the FFTF ISC design analysis report (General Atomics 1995).

The sealing system for the ISC retains its confinement capability when subjected to normal, off-normal, and accident loading conditions because there are no normal or accident conditions that will breach the structural integrity or leaktightness of the ISC.

Handling/Drop – The FFTF cask/container system is analyzed for the DBA drops in Chapter D3.0, and the analysis is summarized in Section D3.4.2.1. Several drop analyses were performed for the ISC. Under normal operations, the ISC is lifted vertically by attaching the lifting fixture to three anchor attachments imbedded in the concrete of the ISC. No side lifting is allowed. In accordance with the critical lift requirements of DOE/RL-92-36, the ISC lift points are designed to lift five times the weight of the cask without exceeding the ultimate stress of the material or three times the weight without exceeding the yield strength of the material, whichever is less. Therefore, the normal ISC lifting and handling loads will generate relatively low stresses in the lift points. The weakest link in the ISC attachment design is the anchor lug, which screws into the anchor bolt of the cask. If a lifting accident occurs whereby a lift point fails, this anchor fails before the ISC concrete embedment. In this manner, the ISC integrity or shielding is not affected.

As discussed in Section D3.4.2.1, the ISC system is shown to survive the following drop accidents:

- 18-ft drop of the CCC into the ISC
- 4-ft sideways drop of the CCC while inside the ISC
- 4-ft ISC drop onto an unyielding surface
- 8-ft ISC drop onto a 1.5-ft thick concrete pad
- 40-in. ISC drop onto a 6-in. pin.
The CCC will be contained within the ISC for the entire duration of this project, so an independent drop of the CCC is not possible. However, the CCC drop analyses are identified to demonstrate that it will survive the drops analyzed for the ISC.

The conclusion of the accident analysis is that the ISC system survives all of the analyzed drops without loss of confinement. The 4-ft ISC drop onto an unyielding surface is the bounding case. The 8-ft ISC drop onto a 1.5-ft thick concrete pad is bounded because the pad yields and absorbs much of the impact energy. Since the 200 Area ISA concrete pad is 1.5-ft thick, an 8-ft drop is tolerable without loss of confinement.

As indicated in the accident analysis in Section D3.4.2.1, TSR controls have been applied to ensure that the ISC system is not taken outside of the analyzed bounds.

**Mobile Crane Fall** – The FFTF cask/container system is analyzed for the DBA in Chapter D3.0, and the analysis is provided in SNF-4790 and summarized in Section D3.4.2.2. The structure of the ISC withstands the induced loads from all mobile crane fall accidents without loss of confinement function. The CCC structural integrity is not compromised.

**Cask Tipover** – The FFTF cask/container system is analyzed for the DBA in Chapter D3.0, and the analysis is summarized in Section D3.4.2.3. Analyses performed for the ISC tipover show that induced stresses and loads are bounded by the 4-ft drop onto an unyielding surface analyzed in Section D3.4.2.1. The CCC structural integrity is not compromised.

**Fuel Rod Rupture** – The accident condition of all fuel rods being ruptured is analyzed for the DBA in Chapter D3.0 and the analysis is summarized in Section D3.4.2.4. The inventory of fission gas and gases produced during 50 years of radioactive decay is expected to be 5.4 gram-atoms (moles)/DFA or 37.8 gram-atoms/ISC. This assumes that all free gases contained within the cladding are released. The analysis assumes the worst-case ISC cavity average temperature of 286 °F. The maximum pressure is determined to be 62 lbs/in² gauge maximum for the CCC worst-case loading of seven DFAs (WHC-SD-FF-ER-100). The design pressure of the CCC is 70 lbs/in² gauge. A finite element stress analysis is also performed for the pressure load of 70 lbs/in² gauge (WHC-SD-FF-DA-077). The CCC structural integrity is not compromised. Each CCC was hydrostatically tested to the design pressure of 105 lbs/in² gauge.

The ISC cavity internal pressure is also calculated for the same conditions (General Atomics 1995). The maximum ISC cavity pressure, based on the maximum cavity average temperature of 286 °F for the ashfall accident case, is determined to be 36.7 lbs/in² gauge, which is the design pressure for the ISC.

**Seismic** – The design basis seismic event for the FFTF cask/container system is analyzed for the DBA in Chapter D3.0, and the analysis is summarized in Section D3.4.2.5. The CCC structural integrity is not compromised, as the ISC does not slide or tip over. All seismically induced loads are considered bounded by the drop impact accelerations.
The ISC is designed to withstand the seismic motions from the DBE conditions (0.25 g) for the 400 Area ISA. The FFTF ISC design analysis report (General Atomics 1995) states that the cask is a very stiff structure and consequently acts as a rigid body during an earthquake. During the maximum anticipated seismic event, the cask will not slide or tip over. Stresses generated by the seismic event are within the allowable defined limit for accident conditions. Supplemental analysis for the 200 Area ISA indicates the ISC will not slide or tip over when the 200 Area ISA seismic inputs of 0.26 g (identified in Table D2-2) are applied. This analysis is documented in HNF-2183, Overturning and Sliding Assessment for the Interim Storage Cask at the 200 Area Interim Storage Area.

**Tornado/Wind** – The DBT event for the FFTF cask/container system is analyzed for the DBA in Chapter D3.0, and the analysis is summarized in Section D3.4.2.6. Wind velocities are bounded by the tornado winds, but the wind missile is also analyzed. A 15-lb., 2x4 traveling at 50 mph only penetrates the concrete surface about 1 in. The tornado winds do not result in sliding or tip over of the cask. There are no induced loads to the CCC as a result of the tornado or wind events.

**Fire** – The conditions of the transportation DBF and peak thermal conditions are analyzed for the DBAs in Chapter D3.0, and the analyses are summarized in Section D3.4.2.7. The fuel cladding temperature limit of 900 °F, the CCC closure seal temperature limit of 500 °F, and the ISC closure seal temperature limit of 700 °F are evaluated.

The fire accident would result in a short-term thermal loading on the ISC. Section D3.4.2.7 summarizes the ISC temperatures in the cask during a fire. Most of the concrete remains below the 350 °F ACI 349-90 accident allowable. For the conditions representing the fire accident, the large mass of concrete helps keep the interior temperatures of the cask relatively low. The maximum liner temperature reaches 309 °F, well within the allowable limit of 495 °F (with argon-inerting gas). At least 17 in. of concrete remain below the 350 °F ACI 349-90 temperature limit defined for accident conditions.

The maximum ISC closure seal temperature for the fire accident scenario reaches 200 °F, well below the allowable limit of 700 °F for the Helicoflex metal seal.

The closure bolt stress generated during the fire accident is 65,032 lbs/in², which is much lower than the 84,000 lbs/in² allowable. Because the bolts do not yield, sealing performance is unaffected by the fire. Therefore, the cask will remain within design limits during a fire.

**D4.4.1.5 Fast Flux Test Facility Interim Storage Cask Controls (Technical Safety Requirements)**. The assumptions associated with the ISC that require TSRs to ensure performance of its safety function are as follows:

- The ISC shall not be lifted more than 8 ft above the ground or storage pad.
• Each ISC shall be placed in a storage array with spacing of at least 24-in. x 44-in. between ISCs measured edge to edge. This specification applies to all ISCs to provide adequate access to the casks and to meet thermal analysis boundary conditions.

• Crane loads other than ISC rigging shall not be operated over a loaded ISC.

• The ISC system may not be lifted over objects taller than 4-ft or containing radioactive materials.

• Only the Manitowoc 4000 150T crane with Model 22 80-ft boom or the Manitowoc 4100 250T crane with Model 27 80-ft boom shall be used to lift the ISC.

• Fire loadings are to be controlled per the fire hazards analysis.

D4.4.2 Rad-Vault System

A 250 kW Training, Research and Isotope Production, General Atomics (TRIGA) experimental research reactor was operated in the 300 Area intermittently from the late 1970s until its last power run in May 1989. The reactor was manufactured by the Gulf General Atomic of San Diego, California, and was used primarily for neutron radiography of FFTF fuel elements and test assemblies. The fuel from the reactor core/pool storage has been removed as part of the decommissioning of the facility. The TRIGA irradiated fuel inventory consists of 99 TRIGA fuel elements and two fuel follower control rods (FFCRs). The TRIGA fuel and the fuel followers will be stored in TRIGA fuel casks and U.S. Department of Transportation (DOT)-6M containers prior to shipment to the 200 Area ISA, and the sealed casks and containers will be placed in concrete Rad-Vaults at the 200 Area ISA.

The concrete Rad-Vault at the ISA will contain six Neutron Radiography Facility (NRF) TRIGA casks and two special DOT-6M containers containing one FFCR each. Each TRIGA cask holds up to 17 fuel elements.

D4.4.2.1 Rad-Vault Safety Function. The Rad-Vault provides passive protection from natural phenomena events that could damage the NRF TRIGA casks or the DOT-6M containers. Based on the results of the Chapter D3.0 consequence analysis, the Rad-Vault would be designated General Service; however, the Rad-Vault has been upgraded to a designation of Safety Significant to accomplish NRC equivalency based on the classification requirements defined in Section D4.2.2. The safety-significant function imposed to accomplish NRC equivalency for important-to-safety Category B is to ensure mitigation of events that could significantly damage storage containers.
Accidents and concerns identified in the Chapter D3.0 analysis and listed in Tables D4-1 and D4-2, as associated with the upgraded safety-significant functions for the Rad-Vault, are as follows:

- **Cask tipover** - Passive design features preclude tipping.

- **Seismic** - Withstand seismic accelerations without tip over or sliding.

- **Tornado/wind** - Withstand DBT winds (excluding DBT missiles) without sliding or tip over. Also withstand design basis wind and wind-driven missiles established for the 200 Area ISA.

- **Fire** - Withstand transportation DBF conditions such that TRIGA casks and DOT-6M/2R containers inside do not lose confinement or exceed temperature limits for fuel cladding or container components.

**D4.4.2.2 Rad-Vault System Description.** The concrete Rad-Vault (CNSI 1992), shown in Figure D2-3, is a low-level radioactive waste storage unit consisting of a right circular (5,000 lbs/in²) concrete cylinder with light steel reinforcement. The Rad-Vault will be placed on compacted gravel at the ISA, and top loaded with the NRF TRIGA casks and DOT-6M containers. The Rad-Vault was constructed in accordance with ACI-301, *Specifications for Structural Concrete for Buildings*, and ACI-318, *Building Code Requirements for Reinforced Concrete*. The Rad-Vault is 111 in. high, with an outer diameter of 114 in. The wall is 18-in. thick steel reinforced concrete, with an 8-in. thick bottom. The equivalent lead shielding is 3.1 in. The empty Rad-Vault weight is 63,400 lbs (43,400-lb body and 20,000-lb lid), and the maximum design weight with payload is 81,400 lbs. The NRF TRIGA storage system's loaded weight is 76,760 lbs. The NRF TRIGA casks and DOT-6M containers and accompanying fuel weigh approximately 13,358 lbs (CNSI 1992).

The Rad-Vault is a commercial item typically used for on-site storage of low-level radioactive waste at commercial nuclear power plants, but is not a licensed container. The Rad-Vault concept provides on-site storage for commercial power plants in the unlikely event that off-site disposal is limited or prohibited. An on-site engineered storage building is not required (CNSI 1992). The design life is not specified by the vendor, but the concrete vault is epoxy coated and has an estimated life, with proper maintenance, of 50 years.

The Rad-Vault is equipped with a removable lid that fully exposes the available internal storage volume. Its mating surface is gasketed with the main container and sloped to prevent rain intrusion. Opposing lifting lugs are interlaced into the steel rebar and welded wire fabric and are cast into the concrete structure. Lift capacity is sufficient to allow an empty Rad-Vault to be lifted and moved by crane or to a transport trailer. The lid must be transported separately. The Rad-Vault is not intended to be lifted loaded or with the lid installed.
Sampling and/or drain capability is provided through a siphon-type arrangement that prevents leakage in and out of the Rad-Vault. Each Rad-Vault is also equipped with a pop-up vent. No hydrogen gas is expected to be vented; however, the pop-up vent and siphon will provide a vented system. The criticality evaluation (WHC-SD-FF-CSER-006) was performed for a flooded vault. There are no safety requirements for the vent or the drain functions. To ensure shielding uniformity and integrity, the Rad-Vault was gamma scanned. The Rad-Vault is painted internally and externally with two coats of epoxy paint.

The fabrication materials for the storage system components for the Rad Vault are as follows:

- ASTM A-615 for bar reinforcement
- ASTM A-185 for welded wire fabric
- ASTM A-36 or equivalent for embedments
- Concrete – properties (including lid): \( f'_c = 5,000 \text{ lbs/in}^2 \) in 28 days, nominal density of \( 145 \pm 5 \text{ lbs/ft}^3 \), a water binder ratio of 0.39 to 0.42, and air entrainment of 5 to 7 percent. ACI-301 specifications were provided for structural concrete. Test reports were included that verified this information. (The wall and lid of the Rad-Vault passed the required gamma scan: all data is part of the Certificate of Compliance package.)

The six NRF TRIGA casks and two DOT-6M containers will be transported to the ISA in three shipments. Three different cask/container configurations in the Rad-Vault will be required to accommodate the fuel transports. The cask configuration in the Rad Vault after the first shipment of two NRF TRIGA casks is shown in Figure D2-7. After the fuel has been transported to the ISA, the impact limiter overpack used during transport will be removed and the NRF TRIGA casks will be lifted by crane into the Rad-Vault. After each placement of containers in the Rad-Vault, the lid will be placed on top by the crane.

Four NRF TRIGA casks will be arranged in the Rad-Vault, as shown in Figure D2-7, after the second shipment of two casks. The third shipment will transfer two additional NRF TRIGA casks and the two DOT-6M containers. The final Rad-Vault configuration of six NRF TRIGA casks and two DOT-6M containers is shown in Figure D2-7. The placement of the casks, containers, and the empty 55-gal drums is based on operational procedures and is not required for criticality safety. The single drum in the center after loading performs a passive function that results in a close packed array, which precludes tipover of the containers within the Rad-Vault.

A nuclear criticality safety evaluation was performed for the transportation of the NRF TRIGA casks to the ISA pad and is included in the NRF TRIGA SARP (WHC-SD-TP-SARP-008). A CSER (WHC-SD-FF-CSER-006) was prepared for the interim
storage of the TRIGA fuel in the ISA to address storage of NRF TRIGA casks and DOT-6M containers in the Rad-Vault. Chapter D6.0 addresses specific criticality aspects of TRIGA fuel storage at the ISA.

D4.4.2.3 Rad-Vault Functional Requirements.

Cask Tipover – The Rad-Vault shall not tip over as the result of natural phenomenon events, acting as a passive barrier and preventing damage to the TRIGA casks or the DOT-6M containers.

Seismic – The Rad-Vault shall withstand any induced stresses resulting from the design basis seismic event with sufficient integrity to act as a passive barrier and prevent damage to the TRIGA casks or the DOT-6M containers.

Tornado/Wind – The Rad-Vault shall withstand any induced stresses resulting from the design basis tornado/wind event with sufficient integrity to act as a passive barrier and prevent damage to the TRIGA casks or the DOT-6M containers.

Fire – The Rad-Vault in the 200 Area storage location shall be capable of withstanding the DBF with sufficient integrity to act as a passive barrier and prevent damage to the TRIGA casks or the DOT-6M and 2R containers.

D4.4.2.4 Rad-Vault System Evaluation. The Rad-Vault is the exterior container of the TRIGA fuel storage system in the 200 Area ISA storage configuration. The Rad-Vault is the only container that is exposed to a majority of the events analyzed. Resistance to the events precludes similar exposure to the TRIGA casks, the DOT-6M containers, and the NRF TRIGA fuel during storage and ensures that the appropriate inner components withstand the design basis events with no loss of confinement. The safety-significant function imposed to accomplish NRC equivalency for important-to-safety Category B, as defined in Section D4.2.2, is to prevent significant damage to the storage containers.

Seismic – The analysis, summarized in Section D3.4.2.5, shows the design of the Rad-Vault is well within the DBE for the site, and seismic stability is assured. The Rad-Vault will not overturn or slide on the soil. The lid lip will not fail, and the lid will remain on the Rad-Vault. The NRF TRIGA casks inside the container will not suffer loss of integrity, and the DOT-6M containers that will be used for the FFCRs are qualified for conditions that are in excess of the DBE.

The flexure and shear stiffnesses of the Rad-Vault are also analyzed in Chapter D3.0 to determine the natural periods, and hence frequencies, of the structure under seismic excitation.

A comparison between the natural horizontal frequency of the Rad-Vault and the response spectra for the Hanford Site safe-shutdown earthquake shows that the Rad-Vault will behave like a rigid body during a seismic event, and dynamic analysis is not required.
**Tornado/Wind** – The Rad-Vault weighs approximately 63,400 lbs. without a load and has a center of gravity located approximately at the geometric center. It is loaded with the six NRF TRIGA casks and two DOT-6M containers. The analysis, summarized in Section D3.4.2.6 and including calculations from the manufacturer (CNSI 1992), shows that an empty Rad-Vault will endure winds and pressure transients in excess of the 200 Area ISA DBT without sliding or tipover.

The Chapter D3.0 analysis indicates that individual NRF TRIGA casks or DOT-6M containers will be exposed to the risk of a tornado for a short time during the crane lift from the overpack to the Rad-Vault by the crane. However, DOE/RL-92-36, Section 3.0, "Critical Lifts," will be used during the lifts, and the critical lift procedure will be prepared to identify any weather-related limits (as required) for the load being transferred. Therefore, the time period in which the NRF TRIGA casks or DOT-6M containers might be exposed to inclement weather conditions is negligible and, therefore, is not analyzed.

The loads resulting from the design basis wind are bounded by the consequences of the DBT discussed above, with the exception of wind-driven missiles. The Chapter D3.0 analysis also determines the penetration of the Rad-Vault by a wind missile. The evaluation concludes that the missile penetration will be minimal.

The Chapter D3.0, Section D3.4.2.6, analysis of the effects of the wind-driven missiles and the bounding evaluation of the effects of the DBT showed that the Rad-Vault is able to withstand any induced stresses without significant damage to the internal containers.

**Fire** – The TRIGA fuel storage system is analyzed in Chapter D3.0 and summarized in Section D3.4.2.7 against the DBF specified for the 10 CFR 71 transportation fire.

The analysis was conducted to determine the thermal effect of excessive heat on the Rad-Vault storage system, and the internal containers and their contents, which results from a combined maximum temperature solar day and fire.

The analysis assumed equilibrium conditions and determined that radiological material releases will not result from exceeding the temperature limits; however, the design specifications for the exterior concrete material of the Rad-Vault can be exceeded for a short period of time and should be evaluated as part of the recovery action, if a fire event occurs.

**D4.4.2.5 Rad-Vault Controls (Technical Safety Requirements).** The assumptions associated with the Rad-Vault that require TSRs to ensure performance of its safety function are as follows:

- The Rad-Vault shall not be moved while loaded or with the lid in place.
- Crane loads other than NRF TRIGA casks, DOT-6M containers, rigging, and the Rad-Vault cover shall not be handled over a loaded Rad-Vault.
• The Rad-Vault cover shall not be handled at a height greater than 12 in. above the top of the Rad-Vault.

• The Rad-Vault lid shall be handled with any appropriate crane within the Hanford Site hoisting and rigging criteria (DOE/RL-92-36) for a critical lift, but no larger than a 250-ton crane.

• The Rad-Vault lid shall be replaced after each activity within the Rad-Vault.

D4.4.3 NRF TRIGA Cask

The fuel from the 300 Area TRIGA experimental research reactor core/pool storage has been removed as part of the decommissioning of the facility. The TRIGA irradiated fuel inventory consists of 99 TRIGA fuel elements and two FFCRs. The standard fuel elements are stored dry in the NRF TRIGA casks prior to transport to the 200 Area ISA.

D4.4.3.1 NRF TRIGA Cask Safety Function. Based on the results of the Chapter D3.0 consequence analysis, the cask would be designated General Service; however, the cask has been upgraded to a designation of Safety Significant to accomplish NRC equivalency based on the classification requirements defined in Section D4.2.2. The safety-significant function imposed is to ensure mitigation of events that could release radioactive material. The second safety function of the NRF TRIGA cask is to prevent an uncontrolled release of the TRIGA fuel material.

Accidents and concerns identified in the Chapter D3.0 analysis and listed in Tables D4-1 and D4-2, as associated with the upgraded safety-significant functions for the NRF TRIGA cask, are as follows:

• Handling/drop – Maintain confinement of radioactive materials after a credible drop.

• Cask tipover – Maintain confinement of radioactive materials after cask tipover.

• Fuel rod rupture – Maintain confinement of radioactive materials after rupture of all the fuel pins.

• Seismic – Withstand seismic accelerations without loss of confinement.

• Tornado/wind – Withstand tornado pressure differential without loss of confinement.

• Fire – Withstand transportation DBF within the Rad-Vault without losing confinement or exceeding temperature limits for fuel cladding or container components.
D4.4.3.2 NRF TRIGA Cask System Description. Each NRF TRIGA cask consists of three main components: an inner aluminum basket to hold fuel elements, a surrounding single-wall stainless steel inner container, and an outer stainless steel and lead composite vessel for shielding. There is also an external impact limiter that is not used during storage (WHC-SD-TP-SARP-008).

The inner basket, shown in Figure D2-4, is a welded assembly of machined aluminum tubing and plate 28.5 in. high. Two 1.0-in. thick circular plates, each machined with a hexagonal array of 18 thru-holes, serve as insertion guides and support members for the TRIGA fuel elements. A third aluminum plate, 2.56 in. thick, serves as the fuel seating structure. The top portion of the central aluminum tube has threads that interface with basket-lifting equipment (WHC-SD-TP-SARP-008).

The surrounding stainless steel inner container, shown in Figure D2-4, contributes to the total shielding capability of the cask. The shell of the inner container is 29.12 in. high, with a 10.25-in. outer diameter and an 8.88-in. inner diameter 304 stainless steel tube section. The shell has a machined O-ring groove at the upper face to accommodate a 9.12-in. elastomer seal. The base of the container is a 9.5-in. diameter 304 stainless steel plate that is 0.5 in. thick, which is welded to the lower end of the shell. The lid is a 9.62-in. diameter 304 stainless steel plate that is 1.0 in. thick. The closure system consists of a locking bar and a single stainless steel eyebolt. The locking bar, with a tapped thread at its upper end, is welded to the center of the interior side of the container base. To close, the eyebolt is inserted through the center of the lid and threaded into the upper portion of the locking bar until the proper seal compression is achieved (WHC-SD-TP-SARP-008).

The outer confinement vessel, shown in Figure D2-4, consists of a 12.0-in. outer diameter 304 stainless steel inner shell with a wall thickness of 0.75 in., which serves as the confinement boundary. The 304 stainless steel outer shell has a 16.0-in. outer diameter with a wall thickness of 0.5 in.

Both shells are welded at the lower end to a 4.0-in. thick by 16.0-in. diameter 304 stainless steel plate. The 1.5-in. thick annulus between the two shells is filled with ASTM B29 lead for shielding. The lid is a 16.0-in. diameter 304 stainless steel plate 3.5 in. thick, which has a machined flange that mates with the top of a seal flange using 12 bolts. The thick center portion of the lid fits closely inside of the seal flange and is provided with an O-ring bore seal. The tolerance between the seal flange and the lid are closer than the bolt tolerance. Therefore, during high loading conditions (e.g., a drop), shear is transferred by bearing to the flange. The total height of the outer vessel with lid is 37.7 in. (WHC-SD-TP-SARP-008).

The weight of the outer vessel is approximately 1,637 lbs. The entire NRF TRIGA cask weighs a total of 1,878 lbs when filled with 18 elements (WHC-SD-TP-SARP-008).
The fabrication materials for the storage system components of the NFR TRIGA cask are as follows:

- Outer vessel – ASTM A-511, 304 stainless steel tubing; ASTM B-29 poured lead; ASTM A-240, 304 stainless steel plate; and ASTM A-540, Grade B23, Class 4 low alloy cap screws.
- Combination Helicoflex O-ring, made of an Inconel spring with a copper jacket, with Viton O-ring.
- Ethylene-propylene O-ring bore seal container seals.
- Inconel X-750, Teflon-coated C-ring.
- Swagelok stainless steel quick disconnect with Viton seal.

The confinement boundary for the NRF TRIGA cask consists of the following vessels, lids, and seals:

- Outer vessel closure lid
- Outer vessel, inner shell
- Outer vessel, seal flange
- Outer vessel bottom plate
- Welds on outer vessel, inner shell
- Helicoflex metallic seals.

The outer confinement vessel closure lid uses an elastomeric O-ring bore seal, located in a perimeter groove of the closure lid step, to establish a $10^{-7}$ scc/sec (air) leak-tight seal within the bore of the cask for initial transport (Figure D2-5). This elastomeric seal has a limited life of approximately 5 years; therefore, a metallic seal is also provided for long-term storage. The Helicoflex metallic seal is located on the surface between the lid and the flange of the outer container. This seal was also tested to $10^{-7}$ scc/sec (air). A leak test port in the lid between the two seals was tested to $10^{-3}$ scc/sec (air). A Viton redundant seal is also provided to facilitate

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1 Teflon is a trademark of E. I. du Pont de Nemours and Company.

2 Swagelok is a trademark of Crawford Fitting Company.
leak testing of the Helicoflex metallic seal. These seals are embedded and secured to the closure lid and interface with the seal flange of the outer container (Figure D2-5). The Helicoflex seal has a design life of 50 years (WHC-SD-TP-SARP-008), and the Viton seal has a design life of 20 years. A seal verification and replacement program, as described in Section D4.4.3.4, is to be implemented.

Two containment welds are located on the inner shell of the outer vessel. Ultrasonic inspection procedures performed in accordance with the ASME Code, Section V (ASME 1989), are used to inspect the top weld. During fabrication, inspections are performed per the ASME Code, Section VIII, Division 1 (ASME 1989). The bottom weld is radiographically inspected in accordance with the same ASME Code. A vendor cask confinement boundary leak test ensures these welds meet the leak-tight criteria per ANSI N14.5. A pressure test at 17.3 lbf/in² was also performed.

Another boundary is provided by the inner container, which uses an elastomeric O-ring imbedded in the step of the lid and a metallic seal around the locking eyebolt (Figure D2-5). This boundary is not leak tested, so credit is not taken for it as a confinement boundary; however, the container will enhance fuel retrieval in the future.

Leak tests can be performed on both the elastomeric O-ring bore seal and the Helicoflex metallic seal by the use of three test ports machined on the top side of the lid (Figure D2-5). The test port at Location 1 penetrates the lid into the cask cavity. The port at Location 2 penetrates the lid into the void space between the bore seal and the metallic seal. The port at Location 3 penetrates the annulus of the Helicoflex metallic seal and the outer Viton seal. Each port is fitted with a quick-connect fitting to facilitate a leak test and a cover plate to protect the fittings and provide shielding equivalent to the lid thickness (WHC-SD-TP-SARP-008). The quick-disconnect fittings are leak tested, and the cover plate’s metallic seal is leak tested to $1.0 \times 10^{-3}$ sec/sec (air) prior to transport.

To accommodate the 10 CFR 72 requirement for redundant seals for commercial fuel, credit must be taken for the Viton O-ring used in conjunction with the Helicoflex metallic seal. Since these elastomer seals have a 20-year design life (less than that of the containers), a program will be implemented to verify seal condition and to replace seals, if necessary. A congruent effort to identify long-life seals for replacement of the original seals will also be implemented.

All stresses evaluated for the NRF TRIGA packaging under normal and accident conditions are maintained below the appropriate ASME Code, Section VIII (ASME 1989) allowables. Design, fabrication, and testing of the NRF TRIGA packaging is in accordance with the requirements of the ASME Code, Section VIII, as required by the appropriate NRC Regulatory Guides. Per Regulatory Guide 7.11, *Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 in. (0.1 m)*, Table 1, the NRF TRIGA package is Category 3. The design follows the criteria from the ASME Code, Section VIII, Division 2, as shown in the NRF TRIGA SARP, Part B, Section 7.0 (WHC-SD-TP-SARP-008).
Fabrication of the NRF TRIGA cask is in accordance with NUREG/CR-3854. All containment welds were radiographically or ultrasonically tested per ASME Code, Section V. During fabrication, inspections and containment leak testing were performed per the ASME Code, Section VIII, Division 1 and ANSI N14.5, respectively.

D4.4.3.3 NRF TRIGA Cask Functional Requirements.

**Confinement** – The NRF TRIGA cask shall withstand potential NPH and accident events with sufficient integrity to provide a confinement boundary for the TRIGA material such that on-site risk acceptance guidelines are not exceeded.

**Handling/Drop** – The NRF TRIGA cask shall withstand any induced stresses resulting from the design basis handling/drop accidents with no significant damage and no uncontrolled release of radioactive material.

**Cask Tipover** – Maintain confinement of radioactive materials after cask tipover.

**Fuel Rod Rupture** – The NRF TRIGA cask shall withstand the internal pressure from rupture of all of the contained fuel elements with no significant damage and no uncontrolled release of radioactive material.

**Seismic** – The NRF TRIGA cask, as an integral component of the Rad-Vault system in the 200 Area storage location, shall withstand any induced stresses resulting from the design basis seismic event with no significant damage and no uncontrolled release of radioactive material.

**Tornado/Wind** – The NRF TRIGA cask, as an integral component of the Rad-Vault system in the 200 Area storage location, shall withstand any induced stresses resulting from the design basis tornado/wind with no significant damage and no uncontrolled release of radioactive material.

**Fire** – The NRF TRIGA cask, as an integral component of the Rad-Vault system in the 200 Area storage location, shall be capable of withstanding the DBF with no significant damage and no uncontrolled release of radioactive material.

D4.4.3.4 NRF TRIGA Cask Evaluation.

**Confinement** – The NRF TRIGA casks were procured with the intent to meet the leaktight requirements of 10 CFR 71 for transportation. The containers are leak tested prior to shipment.

The outer container of the NRF TRIGA cask is the qualified and tested confinement barrier for storage. This container has a HelicoFlex metallic seal for long-term storage integrity, an elastomeric flange seal to facilitate leak testing, and an elastomeric bore seal. The quick
disconnects used for leak testing are also fitted with metallic O-ring seals at the thread interface. The port lids covering the quickdisconnects used for helium leak testing also are provided with single metallic Helicoflex seals.

Helicoflex seals are fabricated using an inner spring covered by an inner lining and a soft outer lining. The Helicoflex seal for the NRF TRIGA has a Nimonic 90 Spring covered by an inner lining of Inconel Alloy 600 and an outer jacket of aluminum. The materials used in the linings are application-dependent and a function of temperature and pressure. The sealing principle of Helicoflex is based, in part, on plastic deformation of the seal's outer lining. The inner helical spring provides an elastic core under bolt preload. Each coil of the spring behaves independently during radial compression. The all-metal design is the reason for its long life.

The NRF TRIGA cask provides multiple barriers. The inner container has an elastomeric seal on the lid and a metallic seal on the clamping eyebolt. The outer container has redundant seals provided by both an elastomeric bore seal O-ring and a Helicoflex combination metallic/elastomeric O-ring seal. Both the bore seal and the Helicoflex seal are leak tested before transport and/or storage.

The closure devices on the NRF TRIGA cask include the closure lid and the three leak test port lids. The closure lid is attached to the cask body with twelve 0.50-in. diameter cap screws, torqued to a minimum of 52±5 ft-lbf. All leak test port lids are attached to the cask lid with six 0.25-in. diameter cap screws, torqued to a minimum of 42±6 in.-lbf.

The NRF TRIGA cask is provided with removable test port covers that permit access to the helium leak test components in the outer container. The container is leak tested upon initial loading according to ANSI N14.5 standards for a maximum leak rate of 1.0×10⁻⁷ sec/sec (air), with the exception that the test port covers are tested to 1×10⁻⁵ sec/sec (air).

The NRF TRIGA cask has a design life of 50 years. All components are stainless steel except the sealed lead annulus and the aluminum basket. There are no neutron poisons. The metallic Helicoflex seal on the outer container has a 50-year design life for this application. The Viton redundant seal has a 20-year design life. The elastomeric bore seal is ethylene propylene and has a 5-year design life. The outer container's elastomeric bore seal is a confinement seal provided to meet transportation requirements. The Helicoflex metallic seal and Viton O-ring are confinement seals provided to accommodate long-term (20-year) storage requirements.

The fuel is stored in untreated air. The stainless steel and aluminum fuel cladding is not susceptible to long-term degradation in unlimited air (ASM 1975 and ASM 1980). There is no corrosion catalyst within the casks. The reactor pool water was deionized, and the stainless steel casks have been passivated.

The hydride fuel has excellent corrosion resistance in water. Bare irradiated fuel specimens have been subjected to a pressurized water environment at 5,700 °F and 1,230 lbs/in² during a 400-hour period in an autoclave (Simnad 1981). The average corrosion rate was
350 mg/cm²/month, accompanied by conversion of the surface layer of the hydride to an adherent oxide film. The maximum extent of corrosion penetration after 400 hours was less than 2 mils. If somehow the cladding was removed from the fuel, the fuel was stored in a corrosive environment (water vapor) within the cask at an elevated temperature, and the cask was pressurized (externally because the fuel would not generate the gas pressure), minimal corrosion will result. For the actual storage configuration: (1) the elements will be dried, ensuring an environment that minimizes corrosion, (2) cladding will be on the fuel, (3) temperatures will not exceed 200 °F, and (4) the resulting maximum pressure will be below the design pressure of 11.2 lbs/in² gauge. Based on the minimal reactivity of the fuel with water, the design storage environment does not require purified dry air or inert gas.

Dr. Simnad’s report (Simnad 1981) and the testing performed on the fuel show there will be no degradation of the fuel, even if the cladding degrades. The fuel itself forms an oxide layer to protect it from corrosion. The environment within the cask is not corrosive for either the cladding or the bare fuel, so the fuel will remain intact. All TRIGA fuel element cladding appears to be intact, as the reactor pool water is not contaminated after 15 years of operation and fuel storage.

Hypothetical degradation of the fuel does not pose unacceptable operational safety considerations with respect to removing the fuel from storage. If repackaging is deemed necessary, it is to be performed in a confinement facility for personnel safety at a yet to-be-determined location.

To accommodate the redundant seal requirement for commercial fuel per 10 CFR 72, credit must be taken for the Viton O-ring used in conjunction with the Helicoflex metallic seal. Since these elastomer seals have a design life less than that of the containers, a program will be implemented to verify seal condition and replace seals, if necessary. An effort to identify long-life seals for replacement of the original seals will also be implemented.

The seal integrity program will include assessment, possible replacement, and verification of seals while maintaining a safe worker environment. It should be noted that due to the relatively low source/dose rates of the TRIGA fuel (maximum cask dose of 92 mrem/h contact, approximately 3 rem shine with cover removed), controlled retrieval and verification may be accomplished in accordance with approved operating procedures in an approved location providing a confinement enclosure. Since the Viton seal has a 20-year design life, this seal integrity program will be developed and included in the authorization basis as required during a later SAR update.

Drop analyses for the NRF TRIGA cask and the DOT-6M container indicate that they retain confinement and design shielding for the normal handling events and the design basis cask/container drops.

The ethylene propylene seal has a tested life of at least 1,000 hours at 275 °F (Parker Hannifin Corporation 1992). The Viton O-ring is rated for a maximum temperature of 400 °F.
The maximum normal temperature for the cask is 161 °F due to solar heating of the Rad-Vault, which is well within the design operating range. The Helicoflex metallic seal, fabricated from Inconel and aluminum, has a maximum operating temperature greater than 480 °F. The maximum normal temperature is 161 °F, which is well within the operating range. The maximum accident temperature to which these seals will be exposed results from the DBF when the interior temperature of the Rad-Vault rises to 177 °F. This temperature does not threaten the integrity of the seals due to the large amount of margin between the design limit and analysis results.

The following aspects of TRIGA fuel storage enhance confinement of the fuel:

- Intact cladding (demonstrated)
- Low burnup (cladding not stressed)
- Low fission gas pressure (cladding not stressed)
- Inner containers (not leak tested)
- Leak tested Swagelok fittings
- Leak tested Helicoflex seal (installed per procedure, tested to $1 \times 10^{-7}$ sec/sec [air]).

These features justify the bolted closure per the approved design. The removable test port covers, as configured, support the seal replacement/demonstration programs required to accommodate 40-year storage.

The inner NRF TRIGA container is fabricated of stainless steel, and the basket is fabricated of aluminum. The 2R inner container of the DOT-6M specification package is constructed of 304L stainless steel, and the support basket is fabricated of aluminum. The fuel cladding is stainless steel or aluminum. There have been reported instances of galvanic corrosion of irradiated aluminum-clad TRIGA fuel stored in stainless steel racks in water. The basket in the NRF TRIGA cask has been specifically designed and fabricated of aluminum to preclude this potential interaction. There are no historical records of stainless steel-clad degradation in aluminum racks. Both types of fuel cladding have been stored in the aluminum fuel storage racks in the 308 reactor, and the reactor core array is also fabricated of aluminum. As such, there is no anticipated interaction between the stainless steel and aluminum metals resulting in deleterious effects on the cladding or container materials.

The TRIGA fuel storage system, composed of the NRF TRIGA casks and DOT-6M containers within the Rad-Vault, can withstand the normal and natural phenomenon conditions anticipated during the 50-year design life of the system. For the DBA identified in Section D3.4.2.2, the Rad-Vault is assumed to fail, allowing damage to the NRF TRIGA casks and DOT-6M containers and resulting in a release of radioactive materials. The off-site and on-site dose consequences are below the release limits and guidelines and also are within limits prescribed in 10 CFR 72 for normal and accident conditions. Although the NRF TRIGA cask and DOT-6M container have the potential to be certified for off-site shipment (they were designed with the intent to meet 10 CFR 71), these containers and the Rad-Vault were all procured as general-service items based on the small radionuclide source term. 10 CFR 72.230
states that casks certified under 10 CFR 71 can be certified as spent fuel storage casks, if they can meet the storage requirements for 20 years. These casks with their metallic seals, in combination with the Rad-Vault, meet the 20-year storage life requirement.

Drop analyses for the NRF TRIGA cask without the impact limiter indicate that the cask retains confinement and design shielding for the normal handling events and the design basis drops. The drop analyses are provided in Section D3.4.2.1.

Under the dynamic conditions of normal transport and hypothetical accidents, the confinement remains sealed.

**Handling/Drop** – During handling operations, any one or all of the NRF TRIGA casks, DOT-6M containers, or the Rad-Vault could be involved in a drop accident that threatens the confinement or results in significant damage to a storage container. A handling/drop analysis is performed in Chapter D3.0 and summarized in Section D3.4.2.1. The analysis demonstrates that under credible drop conditions, the NRF TRIGA casks and DOT-6M containers do not release radioactive material or are not significantly damaged. It is also noted that the side drop analysis bounds a cask tipover event with no loss of confinement. However, to preclude exceeding the design criteria and the analyzed conditions, lifting limits have been imposed and are identified in Chapter D5.0.

**Fuel Rod Rupture** – The NRF TRIGA cask is analyzed in Chapter D3.0, and the analysis is summarized in Section D3.4.2.4. The TRIGA cask design pressure is based on the maximum pressure differential obtained from the 10 CFR 71.71(c)(3) reduced external pressure. The NRF TRIGA cask is pressure tested at 150% of the design pressure.

As noted in the Chapter D3.0 analysis, the low burnup of the fuel produces a small quantity of fission products. Most of the fission product gases are captured within the fuel matrix. The fission product gases within each element would result in negligible pressure in the cask or container, if all the fuel elements are breached within the cask. Byproducts from elastomeric seal degradation will have minimal effect. Therefore, it is concluded that the NRF TRIGA cask pressure will remain less than the design limit.

As noted in the analysis, the NRF TRIGA package contains stainless steel- and aluminum-clad uranium zirconium hydride fuel. This hydride fuel composition and high equilibrium pressure creates a stable, non-hydrogen-producing fuel. There is no generation of hydrogen gas or corresponding pressure rise inside the NRF TRIGA package during normal conditions. The hydride fuel is assumed to be intact and enclosed within the stainless steel or aluminum cladding. Also, the fuel itself is non-pyrophoric and non-water reactive. Therefore, no chemical reaction will occur that produces gas within the NRF TRIGA package. Also, the fuel is air dried with heat lamps prior to loading. This precludes the possibility of radiolytic decomposition of water.
The analysis also indicates that this fuel forms an oxide film that inhibits the loss of hydrogen or degradation of the fuel. Assuming the fuel cladding disintegrates under accident conditions, the temperatures reached will not exceed the temperature required to generate hydrogen.

The analysis concludes that the maximum temperature of the fuel cladding during storage, if experienced inside the cask or container, will result in an insignificant pressure increase in the fuel.

**Seismic** – The NRF TRIGA cask is analyzed as an integral component of the Rad-Vault storage system. The analysis summarized in Section D3.4.2.5 shows the design of the Rad-Vault is well within the DBE for the site, and seismic stability is assured. The Rad-Vault will not overturn or slide, and the NRF TRIGA cask will not suffer loss of integrity or loss of confinement resulting in an uncontrolled release of radioactive material.

**Tornado/Wind** – The NRF TRIGA cask is analyzed for the DBT as an integral part of the Rad-Vault storage system in Chapter D3.0, and the analysis is summarized in Section D3.4.2.6. Wind velocities are bounded by the tornado winds, but the wind missile is also analyzed. A 15-lb., 2x4 traveling at 50 mph only penetrates the concrete surface about 1 in. The tornado winds do not result in sliding or tip over, and there are no induced loads to the cask as a result of the tornado or wind events sufficient to cause a loss of integrity or an uncontrolled release of radioactive material.

**Fire** – Analysis of the DBF is addressed in Chapter D3.0 and summarized in Section D3.4.2.7. The NRF TRIGA cask is analyzed for the DBF as an integral part of the Rad-Vault storage system. An evaluation is provided in Section D4.4.2.4 for the Rad-Vault system containing the NRF TRIGA casks. Fire protection for the NRF TRIGA cask is provided by the Rad-Vault.

**D4.4.3.5 NRF TRIGA Cask Controls (Technical Safety Requirements).** The assumptions associated with the NRF TRIGA cask that require TSRs to ensure performance of its safety function are as follows:

- The maximum handling lift height shall not exceed 109-in. above ground (21 in. above the top of the Rad-Vault with the lid removed).

- The NRF TRIGA cask shall be handled with any appropriate crane within the Hanford Site hoisting and rigging criteria (DOE/RL-92-36) for a critical lift, but no larger than a 250-ton crane.
D4.4.4 DOT-6M and 2R Containers

The fuel from the 300 Area TRIGA experimental research reactor core/pool storage has been removed as part of the decommissioning of the facility. The TRIGA irradiated fuel inventory consists of 99 TRIGA fuel elements and two FFCRs. The FFCRs are stored dry in special DOT-6M containers with 2R inner containers, prior to transport to the 200 Area ISA.

D4.4.4.1 DOT-6M and 2R Container Safety Function. Based on the Chapter D3.0 consequence analysis, the DOT-6M and 2R containers would be designated General Service; however, the containers have been upgraded to a designation of Safety Significant to accomplish NRC equivalency based on the classification requirements defined in Section D4.2.2. The safety-significant function imposed is to ensure the mitigation of events that could release radioactive material. The DOT-6M package and 2R inner container provide a safety function to prevent an uncontrolled release of the FFCR fuel material.

Accidents and concerns identified in the Chapter D3.0 analysis and listed in Tables D4-1 and D4-2, as associated with the upgraded safety-significant functions for the DOT-6M container, are as follows:

- Handling/drop – Provide impact absorption for the 2R container.
- Seismic – Provide impact absorption for the 2R container.
- Fire – Withstand transportation DBF conditions within the Rad-Vault such that the 2R container inside does not lose confinement or exceed temperature limits for fuel cladding or container components.

Accidents and concerns identified in the Chapter D3.0 analysis and listed in Tables D4-1 and D4-2, as associated with the upgraded safety-significant functions for the 2R container, are as follows:

- Handling/drop – Maintain confinement of radioactive materials within the DOT-6M containers after a credible drop.
- Cask tipover – Maintain confinement of radioactive materials after container tipover.
- Fuel rod rupture – Maintain confinement of radioactive materials after rupture of all the fuel pins.
- Seismic – Withstand seismic accelerations without loss of confinement.
Fire – Withstand transportation DBF conditions within the Rad-Vault without losing confinement or exceeding temperature limits for fuel cladding or container components.

D4.4.4.2 DOT-6M and 2R Container System Description. Each of the two DOT-6M containers, shown in Figure D2-6, consists of a 16-gauge 304 stainless steel 110-gal drum that is 23 in. in diameter and 69.25 in. high. An inner (2R) vessel is fabricated from a 304 stainless steel pipe with a 5-in. diameter. A 0.5-in. thick stainless steel plate is welded to the bottom of the pipe. The top of the pipe is sealed by a bolted flange assembly containing one Helicoflex metallic seal and one Viton O-ring. The metal seal is leak tested to $1.0 \times 10^{-7}$ scc/sec (air). The helium test port plug is also leak tested to $1.0 \times 10^{-7}$ scc/sec (air). This inner vessel is partially supported by plywood treated with a fire-retardant coating. Impact-resistant insulating material fills the annulus between the two vessels (WHC-S-0393).

The design and fabrication of the DOT-6M outer and inner container were performed per 49 CFR 178.354 and 49 CFR 178.360, respectively, and the design was approved by DOE. The DOT-6M is a DOT specification packaging approved for off-site shipment as a Type B container. The fabrication of the inner vessel was performed in accordance with the ASME Code, Section VIII, Division 1 (ASME 1989). During fabrication, inspections and containment leak testing were performed per the ASME Code, Section VIII, Division 1, and ANSI N14.5. The weight of the loaded DOT-6M container is approximately 559 lbs.

The fabrication materials for the storage system components for the DOT-6M specification package are as follows:

- Drum – Type 304 or 304 L stainless steel.
- Insulation – Celotex fiber insulation in accordance with ASTM C-208; exterior grade A-C fir plywood; Miracle Type-M Black Magic adhesive; PPG-SPEEDHIDE latex 42-7 fire retardant paint; VIAC Mastic WC-5 weather coating.
- 2R inner container – ASME SA-240, grade 304L plate and sheet; ASME SA-312, grade TP304L seamless or welded pipe; ASME SA-182, grade F304L flanges; ASME SA-276, Type 304L round bar; ASME B18.8.2, Type 18-8 stainless steel dowel pins.
- Helicoflex metallic seal and Viton O-ring.

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3 Celotex is a trademark of the Celotex Corporation.

4 Miracle Type-M Black Magic is a trademark of Miracle Adhesive Corporation.

5 PPG Speedhide is a trademark of PPG Industries, Inc.

6 VIAC Mastic WC-5 is a trademark of VIMASCO Corporation.
When the DOT-6M/2R metal packages were delivered, the 2R materials certification was listed as ASME/ASTM. This equivalency is the result of a long-term joint program by ASTM and ASME. This effort was brought on by the precipitous decrease in nuclear-related construction that made ASME-certified materials almost impossible to obtain in small lots. In many cases, the only difference between ASME steels and identical, more abundant and cheaper ASTM steels has been the testing regime. This availability and cost differential was unnecessarily costly for the nuclear transportation and storage industry. Hence, the goal to bring ASME/ASTM specifications for common steel materials into agreement. The result is an ASME specification with annotation similar to the following: ASME SA-240 specification identical with ASTM A-240 "... except for editorial differences ...

D4.4.4.3 DOT-6M and 2R Container Functional Requirements.

Confinement – The 2R container, as an integral part of the Rad-Vault storage system and as configured with the DOT-6M container, shall withstand stresses induced by potential NPH or accident events at the 200 Area storage location with sufficient integrity to maintain the 2R confinement boundary.

Handling/Drop – The DOT-6M and 2R containers, as integral components of the Rad-Vault storage system in the 200 Area storage location, shall withstand any induced stresses resulting from the design basis handling/drop accidents with no significant damage and no uncontrolled release of radioactive material.

Cask Tipover – Maintain confinement of radioactive materials after cask tipover.

Fuel Rod Rupture – The DOT-6M and 2R container shall withstand the internal pressure from rupture of all of the contained fuel elements with no significant damage and no uncontrolled release of radioactive material.

Seismic – The DOT-6M and 2R containers, as integral components of the Rad-Vault system in the 200 Area storage location, shall withstand any induced stresses resulting from the design basis seismic event with no significant damage and no uncontrolled release of radioactive material.

Tornado/Wind – The DOT-6M and 2R containers, as integral components of the Rad-Vault system in the 200 Area storage location, shall withstand any induced stresses resulting from the design basis tornado/wind with no significant damage and no uncontrolled release of radioactive material.

Fire – The DOT-6M and the 2R inner container, as integral components of the Rad-Vault storage system in the 200 Area storage location, shall be capable of withstanding the DBF and maximum anticipated thermal conditions (including volcanic ashfall) with no significant damage and no uncontrolled release of radioactive material.
D4.4.4.4 DOT-6M and 2R Container System Evaluation.

Confinement – The DOT-6M/2R containers used for storage of the FFCRs provide multiple confinement barriers. The intact cladding is a barrier, and the 2R inner container provides a confinement barrier with the combination metallic and elastomeric O-ring seals. The 2R inner container of the DOT-6M system is the qualified and tested confinement barrier, as discussed in Section D4.4.3.4. This 5-in. diameter pipe vessel has a bolted flange assembly retaining an elastomeric seal and a Helicoflex metallic seal.

Helicoflex seals are fabricated using an inner spring covered by an inner lining and a soft outer lining. The Helicoflex seal for the 2R container has a Nimonic 90 Spring covered by an inner lining of Inconel Alloy 600 and an outer jacket of aluminum. The materials used in the linings are application-dependent and a function of temperature and pressure. The sealing principle of Helicoflex is based, in part, on plastic deformation of the seal’s outer lining. The inner helical spring provides an elastic core under bolt preload. Each coil of the spring behaves independently during radial compression. The all-metal design is the reason for its long life.

Although the 2R container was inerted with helium for leak testing, the FFCR fuel can be stored in untreated air. The stainless steel fuel cladding is not susceptible to long-term degradation in unlimited air (ASM 1975 and ASM 1980). There is no corrosion catalyst within the container. The reactor pool water was deionized.

As stated in Section D4.4.3.4, Dr. Simnad’s report (Simnad 1981) and the testing performed on the fuel show there will be no degradation of the fuel, even if the cladding degrades. The fuel itself forms an oxide layer to protect it from corrosion. The environment within the cask is not corrosive for either the cladding or the bare fuel, so the fuel will remain intact. All TRIGA FFCR cladding was intact, as the reactor pool water was not contaminated after 15 years of operation and fuel storage.

The DOT-6M stainless steel 6M drum provides a barrier for normal conditions only. This barrier contains pressure relief capability via the penetrations covered with pressure-sensitive adhesive filament (WHC-S-0393). The 6M drum does not provide a safety-related confinement function.

To accommodate the redundant seal requirement for commercial fuel per 10 CFR 72, credit must be taken for the Viton O-ring used in conjunction with the Helicoflex metallic seal. Since these elastomer seals have a design life less than that of the containers, a program will be implemented to verify seal condition and replace seals, as necessary. An effort to identify long-life seals for replacement of the original seals will also be implemented, as discussed in Section D4.4.3.4.

The Viton O-ring is rated for a maximum temperature of 400 °F. The maximum normal temperature for the DOT-6M is 161 °F due to solar heating of the Rad-Vault, which is well within the design operating range. The Helicoflex metallic seal, fabricated from Inconel and
aluminum, has a maximum operating temperature greater than 480 °F. The maximum normal temperature is 161 °F, which is well within the operating range. The maximum accident temperature to which these seals will be exposed results from the DBF when the interior temperature of the Rad-Vault rises to 177 °F. This temperature does not threaten the integrity of the seals due to the large amount of margin between the design limit and the analysis results.

The aspects of TRIGA FFCR storage that enhance confinement of the fuel are as follows:

- Intact cladding (demonstrated)
- Low burnup (cladding not stressed)
- Low fission gas pressure (cladding not stressed)
- Leak tested Helicoflex seal (installed per procedure, tested to $1 \times 10^{-7}$ scc/sec [air]).

The DOT-6M outer stainless steel drum can be opened to permit access to the helium leak test port on the top of the 2R inner container. The 2R vessel was leak tested in accordance with ANSI N14.5 standards for a maximum leak rate of $1.0 \times 10^{-7}$ scc/sec (air). This test port closure was also leak tested to $1 \times 10^{-7}$ scc/sec (air).

The FFCR storage system, composed of the DOT-6M containers within the Rad-Vault, can withstand the normal and natural phenomenon conditions anticipated during the design life of the system. Although the DOT-6M container could be certified for off-site shipment, these containers and the Rad-Vault were all procured as general-service items based on the small radionuclide source term. 10 CFR 72.230 states that casks certified under 10 CFR 71 can be certified as spent fuel storage casks, if they can meet the storage requirements for 20 years. These containers with their metallic seals, in combination with the Rad-Vault, meet the 20-year storage life requirement.

Handling/Drop – The DOT-6M and 2R containers are analyzed in Chapter D3.0, and the analysis is summarized in Section D3.4.2.1. Tests were performed on DOT-6M shipping packages (with 2R inner containers) to evaluate their response to a drop onto an unyielding surface and a drop onto a steel punch bar. A detailed observation of the packages following the drop testing showed the security bolts to be tight and no damage to other areas that would indicate the possible loss of confinement. It is also noted that the side drop analysis bounds a cask tipover event with no loss of confinement.

As indicated in the analysis, the only differences between the packages tested and the TRIGA FFCR configuration are the type of seal used in the 2R inner container and the anticipated contents within the packages. The drop-tested packages used an elastomeric O-ring seal, whereas the TRIGA FFCR DOT-6M shipping packages will use a metallic O-ring seal to assure long-term storage requirements. In addition, the drop-tested package used can and bottle internal containers, whereas the TRIGA FFCR DOT-6M shipping packages will contain stainless steel clad fuel rods enclosed in stainless steel pipe. The TRIGA FFCRs are not externally contaminated and the solid hydride fuel itself is enclosed within the stainless steel cladding, which has been demonstrated not to leak.
Based on the results of the analysis, the loaded DOT-6M container has an administrative restriction to equal the NRF TRIGA cask maximum lift height and shall not be handled at a height greater than 21 in. above the top of the Rad-Vault (with the lid removed).

**Fuel Rod Rupture** – Fuel rod rupture is analyzed in Chapter D3.0, and the analysis is summarized in Section D3.4.2.4. As noted in Section D.4.4.3.4, the low burnup of the fuel produces a small quantity of fission products. Most of the fission product gases are captured within the fuel matrix. The fission product gases within each element will result in negligible pressure, if the fuel element is breached within the 2R container.

The Chapter D3.0 analysis indicates that the DOT-6M package contains stainless steel-clad uranium zirconium hydride fuel. This hydride fuel composition and high equilibrium pressure creates a stable, non-hydrogen producing fuel. There is no generation of hydrogen gas or corresponding pressure rise inside the DOT-6M/2R package during normal conditions. The analysis also indicates that this fuel forms an oxide film that inhibits the loss of hydrogen or degradation of the fuel. Assuming the fuel cladding disintegrates under accident conditions, the temperatures reached will not exceed the temperature required to generate hydrogen. The analysis concludes that the maximum temperature of the fuel cladding during storage, if experienced inside the cask or container, results in an insignificant pressure increase in the fuel.

**Seismic** – The DOT-6M and 2R containers are analyzed as an integral component of the Rad-Vault storage system. The analysis in Section D3.4.2.5 shows the design of the Rad-Vault is well within the DBE for the site, and seismic stability is assured. The Rad-Vault will not overturn or slide and the containers will not suffer loss of integrity or an uncontrolled release of radioactive material. Seismic accelerations are bounded by the drop analyses.

**Tornado/Wind** – The DOT-6M and 2R containers are analyzed for the DBT as an integral part of the Rad-Vault storage system in Chapter D3.0, and the analysis is summarized in Section D3.4.2.6. Wind velocities are bounded by the tornado winds, but the wind missile is also analyzed. A 15-lb., 2x4 traveling at 50 mph only penetrates the concrete surface about 1 in. The tornado winds do not result in sliding or tip over, and there are no induced loads to the containers as a result of the tornado or wind events sufficient to cause a loss of integrity or an uncontrolled release of radioactive material.

**Fire** – The DOT-6M and 2R containers are analyzed for the DBT as an integral part of the Rad-Vault storage system in Chapter D3.0, and the analysis is summarized in Section D3.4.2.7. Sufficient fire protection for the DOT-6M and 2R container is provided by the Rad-Vault. Exposure to the DBF or maximum thermal conditions resulted in no loss of integrity or release of radioactive material.
D4.4.4.5 DOT-6M and 2R Container Controls (Technical Safety Requirements). The assumptions associated with the DOT-6M and 2R containers that require TSRs to ensure performance of their safety function are as follows:

- The loaded DOT-6M container shall not be handled at a height greater than 21 in. above the top of the Rad-Vault (with the lid removed). (Administrative restriction to equal NRF TRIGA cask maximum lift height of 109-in.)

- The DOT-6M container shall be handled with any appropriate crane within the Hanford Site hoisting and rigging criteria (DOE/RL-92-36) for a critical lift, but no larger than a 250-ton crane.

D4.4.5 NAC-1 Cask System

The safety-significant functions for individual components of the NAC-1 cask system are discussed as an integral part of the NAC-1 cask system evaluation. The components include the inner canister, the NAC-1 cask, and the ISO shipping/storage container. Specific individual component evaluations are addressed as appropriate.

The 300 Area LWR fuel consists of irradiated commercial LWR fuel provided to the Pacific Northwest National Laboratory Material Characterization Center (MCC). The MCC was responsible for conducting laboratory investigations of nuclear waste forms for the DOE Office of Civilian Radioactive Waste Management program. The 300 Area LWR fuel inventory consists of seven commercial fuel assemblies, five PWR and two BWR fuel assemblies, and 26 individual intact fuel rods and miscellaneous segments.

The 300 Area LWR fuel is to be contained in welded inner canisters (Figure D2-14) placed within NAC-1 or Nuclear Fuel Services (NFS)-4 spent fuel shipping casks that satisfy both on-site transportation and storage requirements. The NAC-1 and NFS-4 casks were fabricated to the same design drawings, but at different times by different corporate owners. NAC purchased the NFS-4 design and fleet of casks from NFS and renamed the cask model NAC-1. Both model casks will be referred to as the model NAC-1 cask throughout this document.

The 300 Area LWR fuel storage system, composed of the NAC-1 cask and inner canister, can withstand the normal and natural phenomenon conditions anticipated during the 50-year design life of the system. As identified in Chapter D3.0 and evaluated in Section D3.4.2, the cask survives all DBAs without loss of confinement and therefore has no radiological release under normal or accident conditions. 10 CFR 72.230 states that casks certified under 10 CFR 71 can be certified as spent fuel storage casks, if demonstrated in a SAR that the casks can meet the storage requirements for 20 years. The NAC-1 cask with its metallic O-ring seals and the inner canister with welded closures exceed the 20-year storage life requirement.
D4.4.5.1 NAC-1 Cask System Safety Function. The NAC-1 cask system provides passive protection from stresses resulting from natural phenomena and accident events that could compromise the integrity of the inner canister and fuel cladding and allow an uncontrolled release of radioactive material in excess of on-site guidelines. The NAC-1 cask is a passive barrier that has been designated Safety Significant for the mitigation of potential consequences resulting from NPH and handling events.

The LWR canister is designated Safety Significant in that it provides a confinement function in the event of pressurization that results from failure of the fuel cladding during the storage life of the LWR fuel. The function is to maintain integrity sufficient to preclude uncontrolled release of radioactive material. It is also designated Safety Class for criticality geometry control, as discussed in Section D4.3.2.

Because of its configuration, the ISO container is designated Safety Significant as part of the NAC-1 cask system in that it provides a passive function in preventing tipping or sliding during accident or NPH events. The function does not require the outer panels of the ISO to remain intact following the design basis event. Neither the ISO container nor the NAC-1 cask have a confinement function.

DBAs identified in the Chapter D3.0 analysis and listed in Tables D4-1 and D4-2, as associated with the safety-significant functions for the NAC-1 cask, are as follows:

- Handling/drop – Provide passive protection of the LWR canister such that it retains structural integrity after a credible drop.
- Mobile crane fall – Provide passive protection of the LWR canister such that it maintains confinement after a crane fall accident.
- Cask tipover – The NAC-1 cask in the ISO container is credited with passive design features that preclude tipping.
- Seismic – Withstand seismic accelerations without tipover or sliding.
- Tornado/wind – Withstand DBT winds (excluding DBT missiles). Also withstand the design basis wind and wind-driven missiles established for the 200 Area ISA, without structural damage to the LWR canister.
- Fire – Withstand transportation DBF conditions such that the LWR canister does not lose confinement or exceed temperature limits for fuel cladding or cask components.
DBAs identified in the Chapter D3.0 analysis and listed in Tables D4-1 and D4-2, as associated with the safety-significant functions for the LWR canister, are as follows:

- **Handling/drop** – Maintain confinement of radioactive materials after a credible drop within NAC-1 cask.
- **Fuel rod rupture** – Maintain confinement of radioactive materials.
- **Seismic** – Withstand seismic accelerations without loss of confinement.
- **Tornado/wind** – Withstand tornado pressure differential without loss of confinement.
- **Fire** – Withstand transportation DBF conditions inside the NAC-1 cask without loss of confinement.

DBAs identified in the Chapter D3.0 analysis and listed in Tables D4-1 and D4-2, as associated with the safety-significant functions for the ISO, are as follows:

- **Seismic** – Provide structural support to the NAC-1 cask to withstand seismic accelerations without tipover or sliding.
- **Cask tipover** – The NAC-1 cask in the ISO container is credited with passive design features that preclude tipping.
- **Tornado/wind** – Provide structural support to the NAC-1 cask to withstand tornado winds without tipover or sliding.

**D4.4.5.2 NAC-1 Cask System Description.** The 300 Area LWR fuel storage system consists of six storage units, each comprised of an inner canister, a NAC-1 cask, and an ISO shipping/storage container. The welded inner canister provides confinement of the intact fuel during storage, as required by 10 CFR 72. The NAC-1 provides a shielding overpack that provides weather, accident, and NPH protection.

The NAC-1 casks, previously licensed by the NRC to transport LWR spent fuel and waste material, will be modified for use at Hanford. Modifications to the cask include the removal and plugging of several valves connected to the cask cavity. The NAC-1 casks are mounted to supports within the ISO container for transportation and will remain in the containers in this configuration during storage at the 200 Area ISA.

Additional modifications to the cask include removal of the anti-rotational lugs within the interior cask cavity to accommodate the spent fuel LWR canister and installation of a solid plug at the impact limiter test port. Also, neutron shield tank pressure relief penetrations have been removed and replaced with threaded solid plugs.
The storage units will be placed by crane alongside each other (1 x 6 array) and evenly spaced 4 ft apart. This spacing is not required for criticality or other safety analysis purposes, but rather for personnel access considerations and maintaining personnel radiation doses ALARA.

The NAC-1 cask is designed and fabricated to the requirements of the ASME Code, Section III, Subsection NB. The external shape of the NAC-1 spent fuel shipping cask approximates a smooth-surface, right circular cylinder that is modified, in that impact limiters protrude radially at both ends (see Figure D2-10). The internal cross-section of the cask cavity is circular. The overall dimensions of the cask include a length of 214 in. (including lid impact limiter) and a maximum cross-sectional envelope diameter of 50 in. The internal cavity of the cask is 178 in. long and 13.5 in. in diameter. The maximum loaded gross weight of the cask, including the maximum fuel and inner canister weight (3,300 lbs), is approximately 47,150 lbs. The principal design features of the cask are the structure, shielding and heat dissipation systems, and the lifting and tie-down systems.

The NAC-1 cask provides shielding and environmental protection of the inner LWR canister. The structures of the NAC-1 cask are constructed of stainless steel. The cask cavity is formed by the inner shell, which is a 14.125-in. outside diameter, 0.3125-in. thick stainless steel shell. The upper end of the shell is welded to the cask cavity flange; the bottom end of the shell is welded to the cask bottom casting. Surrounding the inner shell of the cask is a nominal 6.6875-in. thick annulus of chemical-grade lead (gamma) shielding. The lead is shaped such that approximately 5 in. from the bottom and 30 in. from the top, the thickness is reduced to 5.4375 in. There is an annular void, 5 in. long by 1.25 in. thick, at the bottom end of the gamma shield to allow for any lead expansion during the fire accident. The upper axial shaping is accomplished by reducing the diameter of the outer shell 2.50 in. over a 30-in. length. The lead/steel interface of the inner and outer shell has axial copper fins that are imbedded in the lead and welded to the inner and outer shells to transfer heat across the interface with a minimum temperature gradient (see Figure D2-10).

The outer shell is formed by a 30-in. diameter, 1.25-in. thick stainless steel cylinder reduced to a 27.50-in. diameter at one end. The cask bottom consists of a shaped stainless steel disc with a 30-in. outer dimension and an 8-in. thickness that functions as a gamma shield for the bottom end of the cask. The cavity flange is a stainless steel ring with a 29.75-in. outer dimension, 17.50-in. inner dimension, and an 8.625-in. thickness. The bottom disc end and top flange are welded to the inner and outer shells to form the enclosure for the lead gamma shield.

The primary structure of the NAC-1 cask is the inner shell, lower end casting, upper end casting, bolted closure lid with double O-ring seals, the plugged drain valves and rupture disk penetrations, and the vent/helium fill valve that remains operative. The seals of the cask boundary have the capacity to be continuously monitored or tested during storage via the O-ring seal test port. Credit is taken for protection of the canister by the structural characteristics of the Rad-Vault and the NAC-1 cask to demonstrate no loss of confinement during design basis events and accidents.
The inner shell of the NAC-1 cask is a 0.3125-in. thick, austenitic stainless steel right-circular cylinder with internal dimensions of 13.50 in. in diameter and 178 in. in length. At each end, the shell is welded to large austenitic stainless steel end castings. The lower end casting is 3.50 in. thick, with a central region that is an 8-in. thick frustum of a cone. The upper end cask cavity flange is an 8-in. thick annular casting with a tapered central hole that mates with the closure lid. The inner shell is joined to the austenitic stainless steel upper and lower end castings using circumferential full penetration groove welds. All weld configurations are designed and fabricated to meet the ASME Code, Section III (ASME 1971).

The closure lid is a 7.5-in. thick austenitic stainless steel frustum of a cone with a 2-in. thick flange around its circumference at the larger end. A 0.375-in. x 45° chamfer on the lower outer edge of the lid flange is optional; it has no significant effect on the lid's structural adequacy. The closure lid is retained by six, equally spaced, 1-1/4-7UNCASTM A-320, Grade L43, low alloy steel hex-head capscrews and is sealed at the upper end casting by two Helicoflex metallic O-ring seals.

The two drain valves located in the lower end casting are plugged for the storage configuration.

The pressure relief system located in the upper end casting is plugged for the storage configuration, since no postulated storage pressurization events exceed the design pressure of the cask (65 lbs/in²). Pressures within the NAC-1 cask remain below design pressure limits for both normal and accident conditions.

The valve located in the upper end casting will be used as the helium fill port for inerting the inner cavity of the cask.

There is one other penetration in the wall of the upper end casting. This penetration is used as a test port for the region between the O-ring seals to assure the adequacy of the closure lid seal prior to shipment.

Existing cover plates over cask confinement penetrations will remain in place, as designed, to serve as heat shields during the postulated DBF accident.

Neutron shielding tanks are provided in the 4.5-in. thick annular space formed between the outer shell of the lead gamma shield and a thin stainless steel shell that constitutes the outer cask surface. The neutron shield tanks are not used for the storage configuration at the 200 Area ISA and will contain air.

Upper end shielding is provided by the 7.5-in. thick stainless steel cask lid. The cask lid is a stainless steel casting that also serves as a gamma shield. The lid is a flanged frustum of a cone, 7.5 in. thick with a maximum diameter of 25.5 in. The conical portion of the stainless lid is stepped locally to minimize the gap between the lid and cavity flange, preventing displacement of the lid during the design basis drop. The flanged portion of the lid is a 2-in. thick, 25.5-in.
diameter disc. There are six counterbored clearance holes for the lid bolts and four 1-in. diameter blind-threaded holes for attaching the lid impact limiter. The underside of the lid flange has a groove for the two Helicoflex metallic O-ring seals that seal the cask cavity. The upper face of the cavity flange is machined flat to serve as the sealing surface for the cask lid metallic seals. The cask lid is bolted to the cavity flange by six 1.25-in. diameter hex-head bolts. Bolt heads bear on the cask lid; the shanks penetrate through the lid flange and thread into the cavity flange. Six 1.25-in. diameter holes with HeliCoil thread inserts are provided in the cavity flange for bolting the cask lid to the flange. The bolt heads are drilled for a wire security seal.

The lower end impact limiter structure is a ring that surrounds the cask lower casting, formed from a stainless steel sheet and/or plate welded to the cask outer shell and flange areas. The impact limiter was designed to absorb the energy of the design basis 30-ft free drop. It contains a balsa wood disc placed adjacent to the cask bottom. A 0.125-in. thick sheet of asbestos is placed between the balsa and the cask bottom, and is contained internally. Within the impact limiter, extending radially from the center of the cask, are eight 0.375-in. thick stainless steel gussets. The bottom section of the impact limiter also functions as a pedestal for supporting the cask in the vertical position.

An impact limiter located at the upper end of the cask body is designed to absorb the energy of the design basis side drop accident. The upper impact limiter is a stainless steel-sheathed, balsa-filled ring that surrounds the cask cavity flange. A 0.125-in. sheet of asbestos is positioned between the balsa and cask outer shell, and is contained internally. Within the impact limiter, extending radially from the center of the cask, are eight 0.375-in. thick stainless steel gussets. The upper impact limiter, also a cask support member in the storage configuration, rests in the ISO cradle frame cross-member.

The cask lid is protected from impact by a 12-in. thick, balsa-filled lid impact limiter that covers and overlaps the cask lid and cavity flange. The balsa is enclosed within a 0.109-in. thick stainless steel container. A 0.125-in. sheet of asbestos is positioned between the balsa and the sheet material adjacent to the cask, and is contained internally. The impact limiter is attached to the cask lid by four 1-in. diameter bolts. There are elastomer O-rings in grooves under the heads of the 1-in. bolts and a neoprene gasket on the perimeter of the impact limiter. These impact limiter seals are for weather protection and do not provide a confinement function. Removal of the lid impact limiter allows access to the cask lid.

Lifting devices for the NAC-1 cask are designated as lifting trunnions and rotation trunnions. The lifting trunnions are two 8.625-in. diameter by 3-in. long trunnions located on the perimeter of the upper impact limiter. The cask is lifted by a special handling yoke attached to the two trunnions. The rotation trunnions are two 6.625-in. diameter by 3-in. long trunnions for rotating the cask to and from the horizontal position in the ISO container. The lower trunnions are offset from the cask centerline so that when the cask is lowered into the ISO container, it rotates to a horizontal position as the crane hook is lowered.
Transportation of the NAC-1 cask is in the normal storage configuration with the cask in a horizontal position, secured within the ISO container. Two structural cross-member sections serve as cradles for the cask within the ISO container. The lower (rotation) trunnions of the cask are captured by a notch and clamping plate on the aft cradle. The upper lifting trunnions are also captured by clamping plates, to hold the upper end of the cask as it rests on the impact limiter within a neoprene-lined forward cradle. The lid impact limiter is bolted to the cask lid. The lid impact limiter can be unbolted and rolled away from the cask on a track. The cask and ISO container are lifted by crane from the transport trailer and placed in the storage array on the NAC-1 reinforced concrete storage pad.

The NAC-1 cask contains an inner LWR canister that provides a confinement boundary for the fuel. The inner canister is fabricated from a 12-in. stainless steel pipe with welded base cap and top closures. It has a nominal outside diameter of 12.75 in., with a maximum outside diameter of 13.0 inches at the end cap and a total length of 165.25 in. The bottom cap is machined, so the pipe-to-cap weld is inspectable at the side of the canister. The closure lid contains a penetration that allows the canister to be evacuated, filled with helium, and leak tested to the requirements of ANSI N14.5. Figure D2-14 provides an illustration of the PWR inner canister.

The LWR canisters are designed to provide criticality geometry control such that a loaded canister will have a $k_{eff} < 0.95$ when fully moderated and reflected. A 6.00-in. thick stainless steel gamma shield is provided at each end of the canister.

The design weight of the inner canister is 1,250 lbs. The maximum loaded weight of the canister is not to exceed 3,300 lbs. The maximum internal design pressure of the canister is 75 lbs/in$^2$ gauge, testable to a pressure of 100 lbs/in$^2$ gauge.

A fuel rod container fits within the inner canister to hold individually packaged LWR fuel rods. The container is fabricated from 8-in. schedule 80 stainless steel pipe and is split in two lengthwise and hinged to allow loading. The pipe ends are closed with tack-welded end caps fabricated from 0.125-in. stainless plate. The hinged pipe contains the loose rods within the inner canister. The container fits within a 10-in. schedule 40 pipe also located within the inner canister. Confinement of the radioactive material is provided by the LWR canister and the fuel rod cladding.

The NAC-1 cask is transported and stored within a specially designed ISO shipping container. The ISO container provides weather protection for the cask and does not perform safety functions. The ISO containers were fabricated in two heights, in both 6-ft and 8-ft high models. All containers are painted carbon steel construction.

The 8-ft ISO container, fabricated by Evergreen Heavy Industrial Corporation, is nominally 8 ft high x 8 ft wide x 20 ft long. The container has single full-width doors at each end and a removable roof. The frame of the container is structural steel channel construction with a 0.5-in. (12.7 mm) carbon steel floor. The sides are fabricated of 1.6 and 2.0 mm
corrugated carbon steel. The roof is fabricated of 1.6 mm carbon steel sheet metal supported by 40 x 40 mm angle iron on a 60-cm grid and is slightly pitched to prevent ponding of precipitation. The doors and roof are provided with weather seals. Two of these containers will be used to house NAC-1 casks during fuel transport and storage.

The 6-ft ISO container, fabricated by Adamson Containers Ltd., is 6 ft high x 8 ft wide x 20 ft long. The container has two half-width doors at one end and a removable roof. Materials of construction and dimensions are similar to those used in fabricating the 8-ft containers, except the roof material is 1.4 mm carbon steel sheet metal and the roof is flat. Four of these containers will be used to house NAC-1 casks during fuel transport and storage.

The ISO containers are modified for use at Hanford. The roofs of the containers have been strengthened to resist the design basis snow and ashfall conditions associated with storage at the 200 Area ISA. Additionally, the containers have been repainted white to produce cooler surface temperatures for industrial safety during their storage in direct sunlight at the ISA.

D4.4.5.3 NAC-1 Cask System Functional Requirements. Individual components of the NAC-1 cask system are analyzed in Chapter D3.0 as integral parts of the NAC-1 cask. The evaluation of the DBAs is discussed in Section D4.4.5.4. Specific individual component functional requirements and evaluations are addressed as appropriate.

Confinement – The LWR canister shall provide confinement of the LWR fuel. Confinement, as required by 10 CFR 72, is provided by the intact fuel rod cladding and the LWR canister. The combination of the intact fuel cladding and the LWR canister shall ensure that no uncontrolled release of radioactive material occurs.

Handling/Drop – The NAC-1 cask system shall withstand any induced stresses resulting from the design basis handling/drop accidents with sufficient integrity to act as a passive barrier and prevent damage to the inner canister. The LWR canister, analyzed as an integral part of the NAC-1 cask/container system, shall be capable of withstanding any induced stresses resulting from design basis handling/drop events with sufficient integrity to ensure that confinement of the fuel material is maintained and structural integrity is maintained for criticality geometry control.

Mobile Crane Fall – The NAC-1 cask system shall withstand any induced stresses resulting from accidents involving the crane, with sufficient integrity to act as a passive barrier and prevent damage to the safety-class LWR canister. The NAC-1 cask shall protect the LWR canister to ensure no uncontrolled release of radioactive material occurs.

Cask Tipover – The NAC-1 cask in the ISO container is credited with passive design features that preclude tipping. The NAC-1 cask in the ISO container shall be analyzed to not tip over or slide as a result of NPH or DBAs.

Fuel Rod Rupture – The LWR canister, as an integral component of the NAC-1 cask system, shall withstand the internal pressure resulting from rupture of all contained fuel rods.
without compromising the canister structural integrity required to maintain confinement of the fuel material. The design pressure of the LWR canister is 75 lbs/in² gauge, with a test pressure of 100 lbs/in² gauge. The LWR canister shall ensure no uncontrolled release of radioactive material occurs.

Seismic – The NAC-1 cask system shall withstand any induced stresses resulting from the design basis seismic event with sufficient integrity to act as a passive barrier and prevent damage to the safety-class LWR canister. The NAC-1 cask and the ISO container shall protect the LWR canister to ensure no uncontrolled release of radioactive material occurs. The LWR canister, analyzed as an integral part of the NAC-1 cask/container system, shall be capable of withstanding any induced stresses resulting from the design basis seismic event with sufficient integrity to ensure that confinement of the fuel material is maintained and structural integrity is maintained for criticality geometry control.

Tornado/Wind – The NAC-1 cask system in the 200 Area storage location shall be capable of withstanding the design basis tornado/wind with sufficient integrity to act as a passive barrier and prevent damage to the safety-class LWR canister. The NAC-1 cask and the ISO container shall protect the LWR canister to ensure no uncontrolled release of radioactive material occurs. The LWR canister, analyzed as an integral part of the NAC-1 cask/container system, shall be capable of withstanding the design basis tornado/wind and pressure differential with sufficient integrity to ensure that confinement is maintained.

Fire – The NAC-1 cask system in the 200 Area storage location shall be capable of withstanding the DBF with sufficient integrity to act as a passive barrier and prevent damage to the LWR canister. The LWR canister, as an integral part of the NAC-1 cask system, shall be capable of withstanding the DBF with sufficient integrity to ensure that confinement is maintained. The zircaloy fuel cladding has a maximum cladding temperature limit of 644 °F to ensure failure of the cladding material does not occur. The Helicoflex metallic seals of the NAC-1 cask are limited to an operating temperature of 700 °F. The LWR canister shall ensure no uncontrolled release of radioactive material occurs and structural integrity is maintained for criticality geometry control.

D4.4.5.4 NAC-1 Cask System Evaluation.

Confinement – Confinement of the spent fuel is provided by the intact cladding and by a leak-tight welded LWR canister, as described in Section D4.3.2.2. The welded canister ensures that the inert gas within the canister cavity following leak testing is maintained during the storage of the LWR fuel. The NAC-1 cask provides a credited confinement boundary of the spent fuel for transportation to the 200 Area ISA only. In storage, the NAC-1 cask is not credited for confinement. The LWR canister provides an additional confinement boundary to satisfy the confinement and/or retrieval requirements of 10 CFR 72, even though credit is taken initially for the fuel cladding integrity. The NAC-1 cask is a significant structural system that mitigates most normal, off-normal, and accident loads and conditions. It is this structure that is analyzed for most scenarios in this section. The integrity of the LWR canister remains intact for the bounding
accidents analyzed in Section D3.4.2 for each category due to the design of the NAC-1 cask, and no uncontrolled release of radioactive material occurs. The cask provides a passive safety-significant function with regard to potential consequences resulting from NPH and accident-induced stresses.

**Handling/Drop** - These analyses are addressed in Chapter D3.0, summarized in Section D3.4.2.1, and discussed in the LWR canister safety-class evaluation in Section D4.3.2.4. As indicated in the safety-class evaluation, the calculations considered all components as part of an integral cask/container system.

The cask analyses evaluated to hypothetical accident conditions (10 CFR 71.73) included (1) a free drop of the cask through a distance of 30 ft onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected, followed by (2) a free drop of the cask through a distance of 40 in., in a position for which maximum damage is expected, onto the upper end of a solid, vertical, cylindrical, mild steel bar mounted on an essentially unyielding, horizontal surface.

The postulated damage that is incurred by the NAC-1 cask during the hypothetical 10 CFR 71 accident is minimal and does not diminish the ability of the cask to protect the LWR canister and to maintain the confinement boundary. Any deformation will be highly localized, and the bending stresses in the remainder of the cask will not result in failure of the internal stainless steel shells.

The NAC-1 cask system and the LWR canister are able to withstand any induced stresses from handling/drop events without compromising the integrity or the confinement function of the LWR canister.

**Mobile Crane Fall** - This evaluation is provided in Chapter D3.0 and summarized in Section D3.4.2.2. As indicated, the calculations consider all components as part of an integral cask/containment system. The accident analysis postulates that the crane boom drop may be sufficient to breach cask confinement. The Chapter D3.0 analysis determines that the crane boom will fail before the boom can impart sufficient load on the cask to breach confinement. The analysis also investigates whether the cask could be punctured by the crane or other related lifting and rigging hardware. This analysis determines that although the cask will suffer physical damage, the confinement integrity will be maintained in the maximum crane block lift condition. The maximum crane block height that can be achieved by the crane is 80 ft above the cask.

Analysis of the hydraulic spreader bar ISO lifting fixture drop (Section D3.4.2.2) concludes that bending is the controlling parameter that establishes the requirement for a maximum lift height. Although the cask may suffer physical damage, the confinement integrity of the cask will be maintained by imposing a maximum height condition of 6 ft above the ISO container lid, approximately 14.75 ft above the ground (see Chapter D5.0 for load height restrictions). This height provides sufficient overhead clearance to perform all rigging and hoisting activities at the ISA.
The cask system, with the imposition of the maximum lift height restriction, will withstand any induced stresses from a mobile crane fall without impacting the integrity of the LWR canister.

**Fuel Rod Rupture** – This analysis is addressed in Chapter D3.0 and summarized in Sections D3.4.2.4 and D3.4.2.7. The analyzed hypothetical accident scenario postulates the rupture of all fuel pins of a worst-case hypothetical PWR assembly or the consolidated BWR and PWR individual rods under both normal and fire accident conditions, which causes an increase in cavity pressure that results in possible loss of fuel confinement. The calculation of the maximum normal pressure for the LWR canister assumes a normal cask cavity pressure at the temperature resulting from the hottest summer day (115 °F ambient) added to the pressure increase from fuel rod rupture of all rods. The pressure increase within the cask cavity is due to both a temperature increase from the indoor loading temperature at the 324 Building (68 °F) to the maximum normal cavity storage temperature (372 °F) and 100% rod rupture. The evaluation addresses the fission gas generated during irradiation, initial pressurization within the pins, and the gas within the canister cavity from loading and leak-testing activities.

The pressure within the cavity due to rod rupture at maximum normal conditions is within the design pressure of 75 lbs/in² gauge and the test pressure of 100 lbs/in² gauge established for the LWR canister. Release of this volume of gas into the cask cavity will result in a slightly lower pressure within the cask. Thus, this value bounds the LWR fuel normal conditions within the NAC-I cask cavity, which has a design pressure of 65 lbs/in² gauge that is tested to 100 lbs/in² gauge.

Worst-case pressures are also calculated for 100% rod rupture during the DBF conditions. The resultant pressure at this increased temperature is also well within the design pressures for both the LWR canister and the NAC-I cask.

Although it is not anticipated that free water will be in the cask, the potential impact of moisture within the cask is assessed. The assessment indicates that up to 100 grams of free water could be tolerated, without exceeding the design pressure limits. Since the cavity is vacuum dried, there will not be sufficient water for radiolytic decay to create an explosive atmosphere within the canister or cask cavity.

The NAC-I cask system and the LWR canister are able to withstand any induced stresses from fuel rod rupture events without compromising the confinement function of the LWR canister.

**Seismic** – The seismic evaluation is provided in Chapter D3.0, and the analysis is summarized in Section D3.4.2.5. The NAC-I cask in the ISO container is shown to not tip over or slide as a result of seismic forces. As noted, stresses induced in the NAC-I cask and the LWR canister during the 200 Area ISA DBE are within the cask system design capabilities. The LWR canister safety-class evaluation calculations consider all components to be part of an integral system.
Seismically induced loads are bounded by the drop impact accelerations, the evaluation of which concludes that the cask system is able to withstand any induced stresses from handling/drop events without compromising the confinement function of the LWR canister.

**Tornado/Wind** – The analysis of wind exposure effects of the DBT and wind-driven missiles is provided in Chapter D3.0 and summarized in Section D3.4.2.6. The analysis considered all components as integral parts of an LWR storage system.

The ISO shipping containers are provided in two heights: 6 ft and 8 ft. Since the 8-ft high container has more area exposed to the tornado wind, it was examined as the critical case for wind exposure effects (tipping and sliding). The analyses show that the cask and shipping container will not tip or slide as a result of the DBT (sum of rotational and translational velocities). Tipping of the cask/container is resisted by the force developed as a result of the dead weight of the cask/container. Sliding of the cask/container is resisted by the frictional force between the container and ground.

It is anticipated that the shipping container might sustain considerable damage during the tornado event that results in possible collapse of the side walls and dislocation of the roof sheeting. Thus, the Chapter D3.0 analyses conclude that the shipping container may require inspection and repair following a tornado event, but the cask will endure the wind effects of the DBT. For the wind effects analysis, it is conservatively assumed that the container walls and roof structure remained intact, since this is the condition that imposes the greatest resultant forces relative to tipping and sliding the fuel storage cask.

The LWR canister provides confinement for the stored fuel, but the cask inner shell is the component that must withstand the resultant pressure transient sustained during the DBT event, assuming the ISO container roof and walls collapse. The analysis indicates that the cask inner shell will withstand the pressure transient criteria for the DBT with large margins of safety, and therefore, the NAC-1 cask will endure pressure transient effects of the DBT. The LWR canister design pressure is much greater than the 0.9 lb/in² gauge pressure differential of the DBT.

As noted above, the tornado wind is capable of causing significant physical damage to the shipping container, which will require inspection and repair or possibly repackaging of the stored fuel following a tornado event.

The loads resulting from the design basis wind are bounded by the consequences of the DBT discussed above, with the exception of wind-driven missiles. The wind-driven missiles are smaller and less energetic than the analyzed tornado missile, but have a smaller effective area of impact and require separate evaluation. The consequences of tornado missiles for the NAC-1 cask storage are not required to be analyzed, as noted above. However, the analysis is performed in the NAC-1 SAR (NAC 1990) and is judged to have insufficient energy to perforate, puncture, or cause structural damage to the NAC-1 cask that is sufficient to cause the loss of the confinement function. The analysis, summarized in Section D3.4.2.6, also determines the...
minimum thickness required to prevent perforation by a wind missile. The evaluation concludes that the missile will not puncture the cask water jacket.

The analysis of the effects of the wind-driven missiles and the bounding evaluation of the effects of the DBT (Section D3.4.2.6) show that calculated energies for the design basis wind are well bounded by the cask drop analyses. These analyzed energies are found to be acceptable in Section D3.4.2.1, which concludes that the cask system is able to withstand any induced stresses from handling/drop events without compromising the confinement function of the LWR canister.

Fire – The DBF is analyzed in Chapter D3.0 and summarized in Section D3.4.2.7. The analysis determines the DBF to be bounded by the exposure of the NAC-1 cask and associated ISO shipping container to the 10 CFR 71.73(c)(3) transportation fire.

The analysis of the NAC-1 cask subjected to the transportation fire scenario of 10 CFR 71.73(c)(3) is provided in the original transportation SAR (NFS 1972) for an internal thermal load of 11.5 kW, and in the NAC-1 SAR (NAC 1990) for an internal 750 W thermal load. These analyses both analyze the NAC-1 cask with and without the ISO shipping container and provide bounding conditions for the fire accident scenario for use of the NAC-1 shipping cask for storage of the commercial LWR fuel. The LWR fuel has a maximum internal thermal load of 405 W. The transportation SAR and the NAC-1 SAR analyses are therefore considered sufficient to demonstrate the integrity of the NAC-1 cask for the fire accident conditions. The cask integrity is preserved under both of these conditions for storage of the 300 Area LWR fuel.

The analysis in Section D3.4.2.7 concludes that the NAC-1 cask and the LWR canister will withstand any stresses or temperatures induced as a result of fire, and that the canister structural integrity required to ensure that an uncontrolled release of radioactive material will not occur is not compromised.

D4.4.5.5 NAC-1 Cask System Controls (Technical Safety Requirements). The assumptions associated with the NAC-1 cask and ISO shipping container that require TSRs to ensure performance of their safety function of confinement control are as follows:

- The lift height of 30 ft is not exceeded for the ISO container.
- The crane hook and block lift height of 56 ft is not exceeded.
- The number of individual fuel rods in an NAC-1 cask does not exceed a maximum of 179 PWR rods, or 96.5 BWR rods consolidated with 17 PWR rods.
- The NAC-1 cask in the ISO container shall be handled with only the Manitowoc 4000 150T crane with a Model 22 80-ft boom or the Manitowoc 4100 250T crane with a Model 27 80-ft boom.
D4.5 REFERENCES


ACI-301, 1989, *Specifications for Structural Concrete for Buildings,* American Concrete Institute, Farmington Hills, Michigan.


ACI-349, 1990, *Code Requirements for Nuclear Safety Related Concrete Structures,* American Concrete Institute, Farmington Hills, Michigan.


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Annex D - 200 Area Interim Storage Area


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CHAPTER D5.0

DERIVATION OF TECHNICAL SAFETY REQUIREMENTS
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D5.0 DERIVATION OF TECHNICAL SAFETY REQUIREMENTS

D5.1 INTRODUCTION

A description of the essential features of the Spent Nuclear Fuel (SNF) Project derivation of Technical Safety Requirements (TSRs) is provided in Section 5.1 of the SNF Project Final Safety Analysis Report (FSAR).

D5.2 REQUIREMENTS

The requirements that form the basis for the SNF Project derivation of TSRs are identified in Section 5.2 of the SNF Project FSAR.

D5.3 TECHNICAL SAFETY REQUIREMENTS COVERAGE

The 200 Area Interim Storage Area (ISA) TSRs for the analyzed hazards and accidents are summarized in Table D5-1. This table lists TSR controls in accordance with Chapter D4.0 and the accident analyses in Chapter D3.0. Table D5-1 provides a cross-reference of the respective accident analysis section to the relevant subsections within Section D5.5, where derivation details are arranged by TSR control.

The necessary and sufficient TSR controls are established based on consideration for public safety, significant defense in depth, significant worker safety, and for maintaining radiological consequences below risk evaluation guidelines. SNF Project FSAR Section 5.3 contains details applicable to all SNF Project facilities. Section D5.3.2 contains information specific to the 200 Area ISA, in addition to that provided in Section 5.3.2 of the SNF Project FSAR.

D5.3.1 Criteria

The control selection criteria used for the SNF Project are described in Section 5.3.1 of the SNF Project FSAR.
Table D5-1. Final Safety Analysis Report Section and Technical Safety Requirement Cross Reference. (3 sheets)

<table>
<thead>
<tr>
<th>Final Safety Analysis Report section</th>
<th>Technical Safety Requirement</th>
<th>Control basis</th>
</tr>
</thead>
<tbody>
<tr>
<td>D3.4.2.4.5</td>
<td>There shall be no more than seven fuel assemblies in a CCC.</td>
<td>AC5.8 Source Inventory Receipt Acceptance</td>
</tr>
<tr>
<td>D4.3.1.5</td>
<td>Each CCC shall contain six DFAs in the outer positions, seven DFAs including the center position, or six Ident-69 containers in the outer positions. The loaded CCC can contain a mixture of fuel assemblies and pin containers such that the total is either six or seven, and the number of pin containers is five or less. No partially filled CCCs are permitted.</td>
<td>AC5.8 Source Inventory Receipt Acceptance</td>
</tr>
<tr>
<td>D4.3.1.5</td>
<td>There shall be no more than a total of 1,519 pins in a CCC container to protect the fuel rod rupture pressure calculation assumptions.</td>
<td>AC5.8 Source Inventory Receipt Acceptance</td>
</tr>
<tr>
<td>D6.3.1.1</td>
<td>Only authorized fuel, as identified in Chapters D2.0 and D6.0, can be received at the ISA.</td>
<td>AC5.8 Source Inventory Receipt Acceptance</td>
</tr>
<tr>
<td>D4.3.1.5</td>
<td>Only intact fuel shall be placed into a CCC.</td>
<td>AC5.8 Source Inventory Receipt Acceptance</td>
</tr>
<tr>
<td>D4.3.2.5</td>
<td>Only intact fuel shall be placed into an LWR canister.</td>
<td>AC5.8 Source Inventory Receipt Acceptance</td>
</tr>
<tr>
<td>D3.4.2.4.5</td>
<td>The number of individual fuel rods in the NAC-1 cask does not exceed a maximum of 179 PWR rods or 96.5 BWR rods consolidated with 17 PWR rods.</td>
<td>AC5.8 Source Inventory Receipt Acceptance</td>
</tr>
<tr>
<td>D4.3.2.5</td>
<td>The ISC cannot be lifted more than 8 ft above the ground or storage pad.</td>
<td>AC5.9 Lift Restrictions</td>
</tr>
<tr>
<td>D4.4.1.5</td>
<td>The ISC cannot be lifted over an object that is taller than 4 ft or containing radioactive materials.</td>
<td>AC5.9 Lift Restrictions</td>
</tr>
<tr>
<td>D4.4.1.5</td>
<td>Crane loads other than the ISC rigging shall not be operated over a loaded ISC.</td>
<td>AC5.9 Lift Restrictions</td>
</tr>
<tr>
<td>D4.4.2.5</td>
<td>The Rad-Vault can only be lifted empty with the cover removed.</td>
<td>AC5.9 Lift Restrictions</td>
</tr>
<tr>
<td>D3.4.2.1.5</td>
<td>The Rad-Vault cover shall not be handled at a height greater than 12 in. above the top of the Rad-Vault.</td>
<td>AC5.9 Lift Restrictions</td>
</tr>
</tbody>
</table>
Table D5-1. Final Safety Analysis Report Section and Technical Safety Requirement Cross Reference. (3 sheets)

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<tbody>
<tr>
<td>D4.4.2.5</td>
<td>Crane loads other than NRF TRIGA casks, DOT-6M containers, rigging, and the Rad-Vault cover shall not be handled over a loaded Rad-Vault.</td>
<td>AC5.9 Lift Restrictions</td>
</tr>
<tr>
<td>D3.4.2.1.5</td>
<td>The NRF TRIGA casks and DOT-6M containers cannot be lifted more than 109 in. (approximately 9.1 ft) above the ground or more than 21 in. above the top of the Rad-Vault with the lid removed.</td>
<td>AC5.9 Lift Restrictions</td>
</tr>
<tr>
<td>D3.4.2.1.5</td>
<td>The NAC-1 package cannot be lifted over other objects except the transport trailer.</td>
<td>AC5.9 Lift Restrictions</td>
</tr>
<tr>
<td>D3.4.2.1.5</td>
<td>The NAC-1 package cannot be lifted more than 30-ft from the surface of the 200 Area ISA storage pad.</td>
<td>AC5.9 Lift Restrictions</td>
</tr>
<tr>
<td>D3.4.2.1.5</td>
<td>The ISO spreader bar lift height of 6 ft above the ISO container shall not be exceeded.</td>
<td>AC5.9 Lift Restrictions</td>
</tr>
<tr>
<td>D4.4.5.5</td>
<td>The crane hook and block lift height of 56 ft is not exceeded.</td>
<td>AC5.9 Lift Restrictions</td>
</tr>
<tr>
<td>D3.4.2.5.5</td>
<td>The ISC shall be placed on a concrete pad.</td>
<td>AC5.10 Spacing and Placement</td>
</tr>
<tr>
<td>D3.4.2.6.5</td>
<td>The ISC minimum spacing array area shall be no less than 24 in. by 44 in. between ISCs measured edge to edge.</td>
<td>AC5.10 Spacing and Placement</td>
</tr>
<tr>
<td>D3.4.2.5.5</td>
<td>The Rad-Vault shall be placed on compacted gravel.</td>
<td>AC5.10 Spacing and Placement</td>
</tr>
<tr>
<td>D3.4.2.5.5</td>
<td>The NAC-1 cask shall be placed on a concrete pad.</td>
<td>AC5.10 Spacing and Placement</td>
</tr>
<tr>
<td>D3.4.2.2.5</td>
<td>The ISC and NAC-1 cask is restricted to the use of the Manitowoc 4000 150T crane with Model 22 60-ft boom or Manitowoc 4100 250T crane with Model 27 80-ft boom.</td>
<td>AC5.11 Crane Utilization</td>
</tr>
<tr>
<td>D4.4.1.5</td>
<td>The NRF TRIGA cask, DOT-6M container, and Rad-Vault lid shall be handled with any appropriate crane within the Hanford Site hoisting and rigging criteria (DOE/RL-92-36) for a critical lift, but no larger than a 250-ton crane.</td>
<td>AC5.11 Crane Utilization</td>
</tr>
</tbody>
</table>
### Table D5-1: Final Safety Analysis Report Section and Technical Safety Requirement Cross Reference (3 sheets)

<table>
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</thead>
<tbody>
<tr>
<td>D3.4.2.7.5</td>
<td>Fire loadings are to be controlled per the fire hazard analysis.</td>
<td>AC5.12 Combustible Loading Limits</td>
</tr>
<tr>
<td>D4.4.1.5</td>
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</tr>
</tbody>
</table>

AC = administrative control.  
BWR = boiling water reactor.  
CCC = core component container.  
DFA = driver fuel assembly.  
DOT = U.S. Department of Transportation.  
ISA = interim storage area.  
ISC = interim storage cask.  
NAC = Nuclear Assurance Corporation.  
NRF = Neutron Radiography Facility.  
PWR = pressurized water reactor.  
TRIGA = Training, Research and Isotope Production, General Atomics.
D5.3.2 Safety Structures, Systems, and Components Not Provided with Technical Safety Requirement Coverage

All safety-class and safety-significant structures, systems, and components (SSCs) have been provided with TSR coverage, as identified in this chapter. All safety SSCs are provided with TSR coverage with an Administrative Control (AC) program. There are no Limiting Conditions for Operation (LCOs) identified for the 200 Area ISA.

D5.4 DERIVATION OF FACILITY MODES

D5.4.1 Operational Modes

Modes are used to specify when safety limits, LCOs, or surveillance requirements are required. 200 Area ISA operations include the movement and storage of the casks and inspections and surveillances. The casks and canisters are the only barriers to the uncontrolled release of radioactive material. The safety-class SSCs are passive and protect against criticality.

The normal operational mode for the 200 Area ISA (after movement and storage of the casks) is long-term interim storage, which is the "Operation" mode. There are no clear operational distinctions to provide more than one mode for safety-class SSCs. The hazard presented by the spent fuel in the casks/containers cannot be shut off or reduced by operational changes. Therefore, the 200 Area ISA will have only one mode, which will be designated as the "Operation" mode. The 200 Area ISA is capable of receiving or transferring casks between locations within the facility during this mode. Routine operational and maintenance activities will be performed (e.g., surveillances, inspections, radiological monitoring, and repairs).

D5.4.2 Minimum Staffing Levels

During long-term interim storage, there is no manned shift requirement. Assignment of radiological and maintenance personnel on a periodic basis is coordinated with the Canister Storage Building (CSB) and is considered adequate to perform the inspections, surveillance, repairs, and monitoring necessary to protect the health and safety of the public and the collocated worker.

Qualification training for the assigned personnel is addressed in Chapter D12.0. Emergency response is addressed in Chapter D15.0.
Normal long-term interim storage operations. Determination of the unmanned shifts during normal interim storage operations assumes that routine inspection and surveillance are being conducted by CSB personnel. An inspection or surveillance is considered a low-difficulty task.

Abnormal conditions. Manning during abnormal conditions is the same as for normal/long-term interim storage conditions.

Emergency conditions. The assigned personnel during emergency conditions are needed to ensure an appropriate response to the spectrum of accidents analyzed in Chapter D3.0 (hazardous and non-hazardous). The minimum assigned personnel must make prompt initial notifications and implement initial protective actions to preclude or reduce the exposure of individuals affected by the hazards or unsafe conditions during an emergency.

The CSB facility manager or designee is the building emergency director (BED) with the primary responsibility for assessing the event and implementing protective actions at the 200 Area ISA. The BED also makes on-site notifications, implements emergency management procedures, implements facility emergency plans, classifies events, and controls event response. The BED requests support services, as necessary, to perform administrative functions and the minimum functions required to ensure the health and safety of the public, on-site workers, and the environment.

D5.5 TECHNICAL SAFETY REQUIREMENT DERIVATION

Controls associated with the 200 Area ISA are derived in Table D5-1 and are discussed in this section, except for AC 5.1 through AC 5.6, which will be included in the 200 Area ISA TSR document. AC 5.1, "Purpose," AC 5.2, "Contractor Responsibility," and AC 5.3, "Compliance," define the need for subsequent AC provisions and clarify responsibility according to good management practice. AC 5.4, "Technical Safety Requirement Violation," AC 5.5, "Occurrence Reporting," and AC 5.6, "Organization," satisfy the requirements of U.S. Department of Energy (DOE) Order 5480.22, Technical Safety Requirements, Section 9.e.(5). This DOE Order requires ACs for reporting deviations and identifying staffing requirements. Worker safety program requirements are addressed in other FSAR sections based on design and operations information.

D5.5.1 Limiting Conditions For Operation

During long-term storage operation, there are no identified LCOs at the 200 Area ISA.
D5.5.2 Safety Limits

During long-term storage, there are no identified Safety Limits that could be exceeded at the 200 Area ISA.

D5.5.3 Administrative Controls

AC 5.1 through AC 5.6, in accordance with DOE Order 5480.22, are identified above. Additional controls are defined in the subsections that follow.

D5.5.3.1 AC5.7 – Nuclear Criticality Safety.

Purpose. This control protects the assumptions of the nuclear criticality evaluation in Chapter D6.0 and the accident analyses in Chapter D3.0 to ensure that operations and storage at the 200 Area ISA prevent inadvertent nuclear criticality. The purpose of this AC is to protect features that are relied on to preclude a criticality event at the 200 Area ISA. Key elements of this programmatic AC are derived from contractor procedures that include requirements for criticality safety evaluations, criticality prevention specifications, and criticality training. The criticality controls identified in Chapter D6.0 are as follows:

- Obtain the source inventory from the shipping organization to verify the containers meet the TSR for geometry control. Storage at the 200 Area ISA ensures that geometry is maintained.

- Incorporate lift restrictions as required by AC 5.9.

Derivation Criterion. A nuclear criticality program is required as one of the standard ACs, according to DOE Order 5480.22, Section 9.e.(5).

D5.5.3.2 AC5.8 – Source Inventory Receipt Acceptance.

Purpose. This AC ensures shipments received at the ISA only include hazardous and radiological materials that have been analyzed in this FSAR Annex. The key elements of this AC are as follows:

- Acceptance criteria are established such that fuel type, quantity, and configuration are limited to those authorized in this FSAR Annex. The acceptance criteria shall include the following:
  - There shall be no more than seven fuel assemblies in a core component container (CCC).
Each CCC shall contain six driver fuel assemblies (DFAs) in the outer positions, seven DFAs including the center position, or six Ident-69 containers in the outer positions. The loaded CCC can contain a mixture of fuel assemblies and pin containers such that the total is either six or seven, and the number of pin containers is five or less. No partially filled CCCs are permitted.

There shall be no more than a total of 1,519 pins in a CCC container to protect the fuel rod rupture pressure calculation assumptions.

Only authorized fuel, as identified in Chapters D2.0 and D6.0, can be received at the ISA.

Only intact fuel shall be placed into a CCC.

Only intact fuel shall be placed into an LWR canister.

The number of individual fuel rods in the Nuclear Assurance Corporation (NAC)-1 cask does not exceed a maximum of 179 pressurized water reactor (PWR) rods, or 96.5 boiling water reactor (BWR) rods (95 full-length rods plus 6 quarter-length sealed segments) consolidated with 17 PWR rods.

- Procedures are established to ensure shipments meet acceptance criteria.
- Records are kept, maintained, and available for review that document that the material stored at the ISA meets the acceptance criteria.

**Derivation Criteria.** These controls are necessary to ensure that the ISA does not accept material that has not been analyzed in the design basis.

**D5.5.3.3 AC5.9 – Lift Restrictions.**

**Purpose.** This AC focuses on the importance of the planned operational steps that were identified as critical assumptions in the accident analyses in Chapter D3.0 and delineated in Chapter D4.0. The key elements of this AC, which preclude criticality or loss of confinement accidents, are as follows:

- The interim storage cask (ISC) cannot be lifted more than 8 ft above the ground or storage pad.
- The ISC cannot be lifted over an object that is taller than 4 ft or that contains radioactive materials.
- Crane loads other than the ISC rigging shall not be operated over a loaded ISC.
The Rad-Vault can only be lifted empty with the cover removed.

The Rad-Vault cover shall not be handled at a height greater than 12 in. above the top of the Rad-Vault.

Crane loads other than Neutron Radiography Facility (NRF) Training, Research and Isotope Production, General Atomics (TRIGA) casks, U.S. Department of Transportation (DOT)-6M containers, rigging, and the Rad-Vault cover shall not be handled over a loaded Rad-Vault.

The NRF TRIGA casks and DOT-6M containers cannot be lifted more than 109 in. (approximately 9.1 ft) above the ground or more than 21 in. above the top of the Rad-Vault with the lid removed.

The NAC-1 package cannot be lifted over other objects except the transport trailer.

The NAC-1 package cannot be lifted more than 30-ft from the surface of the 200 Area ISA storage pad.

The crane hook and block lift height of 56 ft is not exceeded.

The ISO spreader bar lift height of 6 ft above the ISO container shall not be exceeded.

All lifts are to be considered critical lifts, as discussed in the Hanford Site hoisting and rigging criteria (DOE/RL-92-36).

**Derivation Criteria.** These controls are necessary to support the assumptions of the safety analysis. The assumptions rely on personnel actions to ensure they remain valid.

**D5.5.3.4 AC5.10 – Spacing and Placement.**

**Purpose.** This AC focuses on the importance of the critical assumptions identified in the accident analyses in Chapter D3.0 and delineated in Chapter D4.0. The key elements of this AC, which preclude loss of confinement accidents, are as follows:

- The ISC shall be placed on a concrete pad.
- The ISC minimum spacing array area shall be no less than 24 in. by 44 in. between ISCs measured edge to edge.
- A method shall be established for indicating proper ISC placement (e.g., painting an area on the storage pad).
• After unloading an ISC from the transport vehicle, verify the ISC is properly positioned on the storage pad and the minimum spacing requirement is met.

• The Rad-Vault shall be placed on compacted gravel.

• Before loading the NRF TRIGA casks and DOT-6M containers, verify the Rad-Vault is properly positioned on the compacted gravel and the minimum and maximum spacing requirements are met.

• The NAC-1 cask shall be placed on a concrete pad.

Derivation Criteria. These controls are necessary to support the assumptions of the safety analysis, design features, and lightning protection for the Rad-Vault.

D5.5.3.5 AC5.11 – Crane Utilization.

Purpose. This AC ensures only cranes and booms that are within the calculational analysis are used. The key elements of this AC, which preclude loss of confinement accidents, are as follows:

• The ISC and NAC-1 cask are restricted to the use of the Manitowoc 4000 150T crane with Model 22 80-ft boom, or the Manitowoc 4100 250T crane with Model 27 80-ft boom.

• The NRF TRIGA cask and DOT-6M container shall be handled with any appropriate crane within the Hanford Site hoisting and rigging criteria (DOE/RL-92-36) for a critical lift, but no larger than a 250-ton crane.

• All lifts are to be considered critical lifts, as discussed in the Hanford Site hoisting and rigging criteria (DOE/RL-92-36).

Derivation Criteria. These controls are necessary to support the assumptions of the safety analysis.

D5.5.3.6 AC5.12 – Combustible Loading Limits.

Purpose. This AC provides protection against exceeding the combustible loadings assumed in the fire hazard analysis. The key elements of this AC, which preclude loss of confinement accidents, are as follows:

• The Fire Hazard Analysis shall be reviewed at least annually to ensure that the program complies with the combustible loading limits (e.g., quantity and location) within the analysis.
The facility's combustible loadings shall be evaluated at least annually and compared against the fire hazard analysis limits. Any necessary adjustments shall be made to comply with the fire hazard analysis loading limitations.

**Derivation Criteria.** These controls are necessary to support the assumptions of the fire hazard analysis.

### D5.6 DESIGN FEATURES

Design features for the 200 Area ISA that, if altered or modified, would have a significant effect on safety are listed below. Descriptions of these design features are provided in Chapter D2.0; the safety functions they perform are described in Chapter D4.0.

#### D5.6.1 Fast Flux Test Facility Spent Nuclear Fuel Core Component Container

Design features for the Fast Flux Test Facility (FFTF) CCC include the following:

- Criticality geometry control
- Corrosion-resistant spent fuel storage
- SNF protection against all credible drops
- Radiological shielding for ALARA (as low as reasonably achievable) protection
- The CCC Helicoflex seal shall have a design temperature of at least 500 °F.

#### D5.6.2 Light Water Reactor Canister

Design features for the light water reactor (LWR) canister include the following:

- Criticality geometry control
- SNF confinement
- Shielding overpack for weather and natural phenomena hazards (NPH) protection
- A gamma shield at the canister end
- The individual loose rods must be retained within the 10-in. diameter sleeve.

#### D5.6.3 Fast Flux Test Facility Interim Storage Cask

Design features for the FFTF ISC include the following:

- SNF secondary confinement
- Passive heat removal
- CCC environmental protection
• Radiological shielding for ALARA protection
• SNF drop protection
• Weather protection with the ISC cover installed.

D5.6.4 NRF TRIGA Cask

Design features for the NRF TRIGA cask include the following:

• Confinement and shielding for the TRIGA fuel
• 20-year storage life requirement of the cask metallic seals
• Seals must meet the leak-tight requirement for transportation.

D5.6.5 DOT-6M and 2R Containers

Design features for the DOT-6M package with the inner 2R container include the following:

• SNF confinement
• Radiological shielding for ALARA protection.

D5.6.6 Rad-Vault

Design features for the Rad-Vault include the following:

• Supplemental shielding for NRF TRIGA casks and DOT-6M containers
• Heat shield for fires
• NPH protection for wind, wind missiles, tornados, and volcanic ashfall
• Weather cover to mitigate snow, dust, sandstorm, and rain.

D5.6.7 NAC-1 Cask and International Standards Organization Shipping Container

Design features for the NAC-1 cask include the following:

• Radiological shielding for ALARA protection
• Protection for the LWR canister
• Protection from stresses from natural phenomena and accident events.
D5.6.8 **At-Grade Storage Pads and Compacted Gravel Structure at 200 Area Interim Storage Area**

Design features for grading include the following:

- Maintain site grading such that the probable maximum precipitation event does not allow water to degrade the storage pad (undercut concrete pad).
- Maintain site grading such that the accumulation of runoff water cannot result in a pad overflow of water.

D5.7 **INTERFACES WITH TECHNICAL SAFETY REQUIREMENTS FROM OTHER FACILITIES**

There are no TSRs of other SNF Project facilities that affect the 200 Area ISA. However, there is interaction of the ISA with existing CSB facilities for surveillance activities and central alarm notification.

D5.8 **REFERENCES**


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CHAPTER D6.0

PREVENTION OF INADVERTENT CRITICALITY
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D6.0 PREVENTION OF INADVERTENT CRITICALITY

D6.1 INTRODUCTION

This chapter provides an evaluation of the potential for a criticality accident in the 200 Area Interim Storage Area (ISA). Controls required to prevent criticality in the ISA are also discussed. Chapter 6.0 of the Spent Nuclear Fuel (SNF) Project Final Safety Analysis Report (FSAR) provides an overall description of the SNF Project criticality prevention program.

The scope of the criticality analysis and control strategy for the ISA includes criticality hazards associated with the handling and storage of SNF at the ISA. Specifically, this includes receipt and unloading of sealed casks from transportation vehicles, movement of casks within the ISA facility, and storage of casks in the facility for a period of up to 40 years. Activities excluded from the scope are: (1) transport of the SNF to the ISA facility, and (2) handling and loading of SNF into casks at the Fast Flux Test Facility (FFTF) and the 300 area prior to transport to the ISA. However, the impact of abnormal events or accidents are considered that may affect the condition or configuration of casks delivered to the ISA (e.g., the criticality hazard associated with receiving a misloaded cask is analyzed). The criticality safety of transportation of SNF to the ISA facility is analyzed in WHC-SD-TP-SARP-008, Safety Analysis Report for Packaging (Onsite) NRF TRIGA Packaging, and WHC-SD-TP-SARP-010, Safety Analysis Report for Packaging (Onsite) Interim Storage Cask.

D6.2 REQUIREMENTS

The requirements that form the basis of criticality prevention are identified in Section 6.2 of the SNF Project FSAR.

This chapter also addresses implementation of requirements to achieve U.S. Nuclear Regulatory Commission (NRC) equivalency in the design of the ISA with respect to criticality prevention based on the double-contingency principle. The double-contingency principle is stated in U.S. Department of Energy (DOE) Order 5480.24, Nuclear Criticality Safety, as follows: "Process designs shall incorporate sufficient factors of safety to require at least two unlikely, independent and concurrent changes in process conditions before a criticality accident is possible." The SNF project has established a $k_{eq}$ limit of 0.95 (including allowance for uncertainty) for the ISA (HNF-SD-SNF-DB-003). As applied in this chapter, contingencies include failure of an engineered feature incorporated in the ISA design or failure of an administrative control affecting criticality safety at the ISA.
D6.3 CRITICALITY CONCERNS

D6.3.1 Fissile Material Form and Inventory

As described in Chapter D2.0, four different dry cask storage containers will be stored in the ISA, including the following:

- Interim Storage Casks (ISC) containing FFTF fuel or experimental targets.
- Six Neutron Radiography Facility (NRF) TRIGA casks containing Training, Research and Isotope Production, General Atomics (TRIGA) fuel.
- Two U.S. Department of Transportation (DOT)-6M casks containing the TRIGA fuel follower control rods (FFCRs).
- Nuclear Assurance Corporation (NAC)-I casks containing 300 Area light water reactor (LWR) fuel.

The layout of casks on storage pads and within the Rad-Vault at the ISA facility is described in Section D2.3 and illustrated in Figure D2-1. The design and specifications of the casks themselves are described in Section D2.6. Details of the fuel design and inventory stored in the casks are found in Section D2.5. For the purposes of criticality safety, the fissile material form and contents of the fuel stored in the casks are summarized in the subsections that follow.

D6.3.1.1 Fast Flux Test Facility Fuel. Several different types of fuel pins were analyzed in the criticality safety evaluation reports (CSERs) prepared for the storage of FFTF fuel (see Section D2.5.1.1 for details), as follows:

- Mixed oxide (MOX) fuel pins - The MOX fuel pins contain uranium dioxide and plutonium dioxide in pellet form clad with stainless steel. Slight variations in the enrichment of the fuel exist within the inventory. However, enrichment is bounded by a maximum 29.28 wt% Pu (239 and 241) and a minimum 11.63 wt% Pu-240.

- Metal fuel pins - The metal fuel pins contain a sodium-bonded uranium/zirconium alloy clad with stainless steel. The metal fuel is of two enrichments: 31 wt% U-235 or 32.4 wt% U-235. Note: Storage of metal fuel is not currently authorized at the ISA (see Section D2.5.1.1), but is included in this chapter for completeness since a criticality analysis has already been performed for the metal fuel.

- Certain experimental fuel pins and repackaged fuel debris - This fuel is classified as equivalent to MOX fuel (see Section D2.5.1.1 for details). Note: Storage of experimental fuel debris is not currently authorized at the ISA (see Section D2.5.1.1),
but is included in this chapter for completeness since a criticality analysis has already been performed for the fuel debris.

Note: MOX and metal fuel were analyzed for criticality safety at the ISA. Other FFTF fuel types (e.g., those contained in certain "test fuel assemblies") (see Section D2.5.1.1) cannot be accepted at the 200 Area ISA without further criticality analysis.

The FFTF CSERs analyzed core component container (CCC) tubes loaded with a combination of the following: (Note: storage of metal fuel and experimental fuel debris are not yet authorized for the ISA. However, storage configurations of these fuels that are acceptable from a criticality safety standpoint are discussed here since they have already been analyzed in the FFTF CSERs.)

- Driver fuel assemblies (DFAs) consisting of a hexagonal array of 217 MOX fuel pins.
- Ident-69 containers with varying numbers of MOX fuel pins (up to 217).
- MOX fuel test assemblies consisting of a hexagonal array of 217 or 169 fuel pins.
- Metal fuel test assemblies consisting of a hexagonal array of 169 metal fuel pins.
- Modified Ident-69s containing returned experimental fuel pins and repacked fuel pin debris.

FFTF fuel will be stored in a given ISC subject to the following restrictions:

- No Ident-69, metal assemblies, or modified Ident-69s with repackaged fuel are allowed in the center CCC position. (Note: Ident-69 containers are too long to fit in the center position without modification.)
- The center tube of the CCC must be empty if there are six Ident-69s.
- Mixtures of metal assemblies and Ident-69s are not allowed.
- Modified Ident-69s must not contain over 4 kg of MOX fuel and the enrichment must be <31 wt% Pu.

These restrictions were developed in the FFTF CSERs to ensure safe subcritical limits are not violated during handling and loading of the ISCs at FFTF.

Numerous possibilities exist for the actual loading of the six CCC tubes with the above tube loading options. The $k_{\text{eff}}$ of various bounding configurations for the CCC tube is analyzed in Section D6.3.2.
The fissile material inventory for each fuel type and storage configuration within a CCC storage tube is shown in Table D6-1. This inventory represents the upper bound on plutonium and uranium enrichment.

**D6.3.1.2 NRF TRIGA Fuel.** As described in Section D2.5.1.2, two types of fuel elements will be stored in the six NRF TRIGA casks: (1) stainless steel-clad fuel elements with ceramic zirconium hydride/uranium fuel having 8.5 wt% uranium content with 20% enrichment, and (2) aluminum-clad fuel elements of a similar specification. Two FFCRs will also be stored in the Rad-Vault in DOT-6M casks. The uranium content (shown in Table D2-8) varies slightly by fuel element, but does not exceed 41 g of U-235.

There are a total of 66 aluminum-clad fuel elements, 33 stainless steel-clad fuel elements, and 2 FFCR elements. The total fissile material content of the Rad-Vault will not exceed 4 kg of U-235.

**D6.3.1.3 Commercial Light Water Reactor Fuel.** The fuel stored in the NAC-1 casks is described in detail in Section D2.5.3. Six casks will be used to store (1) five pressurized water reactor (PWR) assemblies and a special loose pin container with pins from two boiling water reactor (BWR) assemblies, and (2) twelve Segmented Rod Program segmented rods from several commercial power reactors. The fuel rods consist primarily of uranium oxide fuel clad in zircaloy, with U-235 enrichment varying from 2.45 to 3.6 wt%. From Table D2-11 the total U-235 content of all the fuel stored in the NAC-1 casks is calculated to be less than 63 kg.

### Table D6-1. Fuel Inventories.

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Core Component Container Tube Configuration (mass in grams)</th>
<th>Metal fuel assembly</th>
<th>Mixed oxide fuel in Ident-69 pin container</th>
<th>Experimental fuel in modified Ident-69 pin container</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu-239</td>
<td>Mixed oxide fuel in driver fuel assemblies and mixed oxide test assemblies</td>
<td>8,421</td>
<td>9,042</td>
<td>953</td>
</tr>
<tr>
<td>Pu-240</td>
<td></td>
<td>1,152</td>
<td>1,237</td>
<td>130</td>
</tr>
<tr>
<td>Pu-241</td>
<td></td>
<td>100</td>
<td>108</td>
<td>11</td>
</tr>
<tr>
<td>Pu-242</td>
<td></td>
<td>19</td>
<td>20</td>
<td>2</td>
</tr>
<tr>
<td>U-235</td>
<td></td>
<td>47</td>
<td>13,971</td>
<td>50</td>
</tr>
<tr>
<td>U-238</td>
<td></td>
<td>23,274</td>
<td>29,102</td>
<td>24,990</td>
</tr>
<tr>
<td>Am-241</td>
<td></td>
<td>42</td>
<td>45</td>
<td>5</td>
</tr>
</tbody>
</table>
D6.3.1.4 Total Interim Storage Area Inventory. The total ISA inventory consists of fissionable material found in 210 FFTF DFAs, 65 FFTF test DFAs, and 11 FFTF test fuel assemblies (those classified as equivalent to MOX fuel), 5 LWR fuel assemblies, 2 FFCRs, 118 loose LWR pins, and 101 TRIGA fuel pins. The total FFTF fuel inventory was inferred from the total Pu and U mass in FFTF fuel listed in Section D2.5.1.1. The fissionable material totals in Table D6-2 are upper bounds on plutonium and uranium enrichment used in the criticality analysis and may differ slightly from actual inventories due to variations in enrichment.

Table D6-2. Total Interim Storage Area Inventories.

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Two interim storage cask pads</th>
<th>NAC-1 pad</th>
<th>Rad-Vault</th>
<th>Total (kg)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu-239</td>
<td>2,085</td>
<td></td>
<td></td>
<td>2,085</td>
</tr>
<tr>
<td>Pu-240</td>
<td>285</td>
<td></td>
<td></td>
<td>285</td>
</tr>
<tr>
<td>Pu-241</td>
<td>24</td>
<td></td>
<td></td>
<td>24</td>
</tr>
<tr>
<td>Pu-242</td>
<td>4.7</td>
<td></td>
<td></td>
<td>4.7</td>
</tr>
<tr>
<td>U-235</td>
<td>95</td>
<td>63</td>
<td>4</td>
<td>162</td>
</tr>
<tr>
<td>U-238</td>
<td>5,937</td>
<td>2,230</td>
<td>20</td>
<td>8,187</td>
</tr>
<tr>
<td>Am-241</td>
<td>10</td>
<td></td>
<td></td>
<td>10</td>
</tr>
</tbody>
</table>

D6.3.2 Criticality Hazards

Table D6-3 identifies criticality hazards associated with handling and storage of the SNF. These hazards apply to all four types of casks, with exceptions as noted.

Changes in $k_{\text{eff}}$ of the casks or groups of casks due to the above hazards are analyzed in Section D6.3.3. The $k_{\text{eff}}$ analysis includes the effect of multiple hazardous conditions from Table D6-3 occurring at the same time. For example, the effect of flooding of a cask in which fuel has disintegrated due to a corrosion phenomenon called hot-cell rot is considered for the ISCs.
Table D6-3. Criticality Hazards Evaluation. (2 sheets)

<table>
<thead>
<tr>
<th>Event Label</th>
<th>Description</th>
<th>Cause</th>
<th>Hazardous effect</th>
<th>Probability</th>
<th>( k_{\text{eff}} ) analyzed?</th>
</tr>
</thead>
<tbody>
<tr>
<td>Interim storage area flooding</td>
<td>Large scale flooding in the ISA</td>
<td>Natural phenomenon such as rain or snow melt</td>
<td>Water introduction into cask(s) increases ( k_{\text{eff}} ) in cask(s)</td>
<td>Not credible, ISA site is above the flood plain.</td>
<td>Yes, can be inferred from the effects of single cask flooding.</td>
</tr>
<tr>
<td>Cask flooding</td>
<td>Localized flooding into one or several casks</td>
<td>Fire hydrant, rain, snow melt, fire suppression</td>
<td>Water introduction into cask(s) increases ( k_{\text{eff}} ) in cask(s)</td>
<td>Not credible for ISCs. Very unlikely for TRIGA and LWR fuel, due to multiple barriers against water introduction.</td>
<td>Yes</td>
</tr>
<tr>
<td>Hot-cell rot</td>
<td>Fuel pins and/or assembly ducts disintegrate over time</td>
<td>Stress corrosion cracking under adverse conditions within cask</td>
<td>Fuel relocated to a more reactive geometry</td>
<td>Anticipated for FFTF fuel, corrosion effects not well known so fuel pin and assembly duct failure is assumed. Not credible for TRIGA and LWR fuel.</td>
<td>Yes</td>
</tr>
<tr>
<td>Tipping/dropping</td>
<td>Cask dropped or crushed during handling</td>
<td>Crane or transport vehicle failure, operational error while handling casks</td>
<td>Fuel relocated to a more reactive geometry</td>
<td>Unlikely (1)</td>
<td>Yes</td>
</tr>
<tr>
<td>Neutron interaction between casks</td>
<td>Casks are arranged too close on a pad or there are too many casks</td>
<td>Administrative error in cask placement</td>
<td>Increased ( k_{\text{eff}} ) due to neutron interaction between casks</td>
<td>Unlikely</td>
<td>Yes, infinite, close packed cask arrays were analyzed.</td>
</tr>
<tr>
<td>Seismic</td>
<td>DBA seismic event tips cask(s) over</td>
<td>N/A</td>
<td>Cask tips. Fuel relocated to a more reactive geometry</td>
<td>Not credible, tipping of cask shown to be not credible by seismic analysis.</td>
<td>Yes</td>
</tr>
</tbody>
</table>
Table D6-3. Criticality Hazards Evaluation. (2 sheets)

<table>
<thead>
<tr>
<th>Event Label</th>
<th>Description</th>
<th>Cause</th>
<th>Hazardous effect</th>
<th>Probability</th>
<th>$k_{\text{eff}}$ analyzed?</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fire</td>
<td>Fire within the ISA perimeter</td>
<td>Range fire, brush accumulation within ISA perimeter, vehicle fuel spill</td>
<td>Increased temperature changes $k_{\text{eff}}$ of casks</td>
<td>Not credible. Worst-case fire has been analyzed and shown not to have a significant effect on fuel temperature (Section D3.4.2.7).</td>
<td>No</td>
</tr>
<tr>
<td>Cask misloading</td>
<td>Cask is loaded with incorrect fuel assembly(s)</td>
<td>Administrative error when loading casks at the facility of origin</td>
<td>Increased $k_{\text{eff}}$ due to incorrect fuel arrangement</td>
<td>Unlikely, due to administrative controls on cask loading before transfer to the ISA.</td>
<td>Yes</td>
</tr>
</tbody>
</table>

(1) Water intrusion into an ISC was judged to be not credible due to the multiple barriers that exist to prevent this hazard. For example, the ISC has an environmental cover to prevent snow and rain intrusion, and two metallic O-ring seals that must fail to allow flooding. The ISC annulus must fill with water before challenging the CCC seals. The CCC inside the ISC has one O-ring seal plus the inner metal seal that must fail. Water intrusion into the TRIGA casks was considered very unlikely. The TRIGA casks have a Helicoflex metallic seal for long-term storage integrity. Also, the Rad-Vault in which the TRIGA casks are stored has a lid designed to prevent rain intrusion. The NAC-1 casks are within an ISO container, which provides weather protection. The LWR canister is welded closed.

(2) Cladding failure at a rate anticipated for commercial LWR fuel has been shown to have a negligible effect on the LWR fuel in SNF-4875, 1999, Criticality Evaluation for Long Term Storage of Light Water Reactor Fuel in NAC-1 Casks. The zircaloy cladding will not suffer significant corrosion at the storage temperatures in the casks. A small percentage of pins might have cladding with breaches as a result of irradiation. These breaches could result in partial conversion of $\text{UO}_2$ to $\text{U}_3\text{O}_8$ in the affected pins, resulting in swelling and pin breakage. However, the number of pins affected is too small to significantly affect the $k_{\text{eff}}$ of the assemblies. TRIGA fuel is a metallic uranium hydroxide that is not susceptible to corrosion in water or air. The hot-cell rot phenomenon is unique to the FFTF fuel.

(3) The worst-case drop or crane collision has been analyzed for the ISCs in Chapter D3.0 and shown not to challenge the integrity of the CCC or ISA. However, the structural integrity of the fuel assemblies is not known and is assumed to fail in a drop. Also, hot-cell rot may occur in the FFTF fuel assemblies, so tipping the ISC could also result in fuel geometry changes. Crushing of the Rad-Vault and TRIGA casks has not been demonstrated incredible for the catastrophic impact of a crane failure. Effects of this event are analyzed in Section D6.3.3. The structural integrity of the LWR canisters and the LWR fuel rods will not be affected by the worst-case cask drop or natural phenomena hazard events, as discussed in Chapter D3.0.

(4) Misloading of Ident-69 containers into the center tube of a CCC was not considered credible, as the Ident-69 is too long to fit in the center tube without physical modification.

DBA = design basis accident.  
FFTF = Fast Flux Test Facility.  
NAC = Nuclear Assurance Corporation.  
ISA = interim storage area.  
SARP = safety analysis report for packaging.  
TRIGA = Training, Research and Isotope Production, General Atomics.  
LWR = light water reactor.
D6.3.3 Analysis of Hazardous Conditions

Evaluations of neutron multiplication factors ($k_{ef}$) for hazardous conditions affecting SNF stored at the ISA have been performed in several CSERs that are summarized in this section. These documents were prepared several years prior to development of the SNF Project safety analysis report. The existing CSERs formed the authorization basis for transportation and storage of SNF at the FFTF 400 Area ISA. These historical CSERs employed several different methods for calculating bias and uncertainty in $k_{ef}$ and determining the upper bound on $k_{ef}$, which are summarized in this section. For comparison to the safe subcritical limit of 0.95, the uncertainty and bias (at the 95% confidence level) reported by the CSERs have been added to the values of $k_{ef}$ calculated in the CSERs, as described in Section 6.3 of the SNF Project FSAR. These "upper bound" values of $k_{ef}$ have been reported in the discussion of analysis results below (as opposed to the "unadjusted" value of $k_{ef}$ calculated by the analysis models in the CSERs).

D6.3.3.1 Fast Flux Test Facility Fuel. The $k_{ef}$ of numerous normal and accident conditions for FFTF fuel stored in ISCs have been analyzed in the CSERs prepared for the storage of FFTF spent fuel (WHC-SD-FV792-DA-004; WHC-SD-FF-CSER-002 and WHC-SD-FF-CSER-004, Revs. 1, 1-A, and 1-B). The analyses in these documents are summarized in Sections D6.3.3.1.1 and D6.3.3.1.2.

D6.3.3.1.1 Analysis Models.

Criticality Safety Evaluation of Ident-69s in the Core Component Container

WHC-SD-FV792-DA-004, Criticality Safety Evaluation of Ident 69s in Core Component Container, presents an analysis of $k_{ef}$ in ISCs loaded with six Ident-69 pin containers, with the center tube of the CCC empty. The analysis was performed with the MCNP Monte Carlo program, version 3B. A parametric analysis was performed on the Ident-69 configuration for a flooded CCC. The following parameters were varied:

- Total number of pins
- Water density
- Square array vs. hexagonal array
- Number of CCC tubes loaded.

From the parametric analysis, the most reactive configuration for the Ident-69 was determined to be 97 pins in a square array. This analysis did not consider the effects of hot-cell rot. All cases relevant to ISA storage assume intact fuel pins. It is known that FFTF fuel lattices with a pin pitch less than 0.90 in. are under-moderated, thus the Ident-69 $k_{ef}$ at the optimal pin pitch of ~.5 in. corresponding to 97 pins was found to be most reactive when the CCC tubes were fully flooded with water. The optimal 97 pin configuration provides an upper bound on the $k_{ef}$ of all flooded Ident-69 configurations. This configuration was used in the later CSERs (WHC-SD-FF-CSER-002; and WHC-SD-FF-CSER-004, Revs. 1, 1-A, and 1-B) to determine $k_{ef}$ for flooded configurations reported in Table D6-4.
### Table D6-4. Interim Storage Cask $k_{\text{eff}}$ Analysis Summary Results. (3 sheets)

<table>
<thead>
<tr>
<th>CCC loading$^{(1)}$</th>
<th>Hazardous Condition(s) present</th>
<th>Upper bound on $k_{\text{eff}}^{(3)}$</th>
<th>Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>7 DFAs</td>
<td>Misloaded Flooded$^{(1)}$ Tipped Hot-cell rot</td>
<td>0.453</td>
<td>WHC-SD-FF-CSER-004, Rev. 0, pg. 12</td>
</tr>
<tr>
<td>7 DFAs</td>
<td>X</td>
<td>0.9494</td>
<td>WHC-SD-FF-CSER-004, Rev. 0, pg. 12</td>
</tr>
<tr>
<td>7 DFAs</td>
<td>X</td>
<td>0.7034</td>
<td>WHC-SD-FF-CSER-002</td>
</tr>
<tr>
<td>7 DFAs</td>
<td>X</td>
<td>0.9905</td>
<td>WHC-SD-FF-CSER-004, Rev. 0, pg. 14 (duct failure assumed - allows fuel compaction to 13 in.)</td>
</tr>
<tr>
<td>7 DFAs</td>
<td>X</td>
<td>1.1553</td>
<td>WHC-SD-FF-CSER-004, Rev. 0, pg. 14 (duct failure assumed - fuel height 36 in.)</td>
</tr>
<tr>
<td>7 Ident-69s</td>
<td>X</td>
<td>0.4530</td>
<td>Bounded by 7 dry DFAs.</td>
</tr>
<tr>
<td>6 Ident-69s (center tube empty)</td>
<td>X</td>
<td>0.4530</td>
<td>Bounded by 7 dry DFAs.$^{(4)}$</td>
</tr>
<tr>
<td>6 Ident-69s (center tube empty)</td>
<td>X</td>
<td>0.8859</td>
<td>WHC-SD-FF-CSER-004, Rev. 0, pg. 17</td>
</tr>
<tr>
<td>6 Ident-69s (center tube empty)</td>
<td>X</td>
<td>Bounded by 7 DFAs with hot-cell rot.$^{(5)}$</td>
<td></td>
</tr>
<tr>
<td>6 Ident-69s and center tube with DFA$^{(5)}$</td>
<td>X</td>
<td>0.9693</td>
<td>WHC-SD-FF-CSER-004, Rev. 1, App. C, pg. 43</td>
</tr>
<tr>
<td>6 Ident-69s and center tube with DFA$^{(5)}$</td>
<td>X</td>
<td>0.9494</td>
<td>Bounded by 7 dry DFAs with hot-cell rot.$^{(6)}$</td>
</tr>
<tr>
<td>5 Ident-69s mixed with DFAs filling remaining CCC tubes, with DFA in center tube</td>
<td>X</td>
<td>0.9498</td>
<td>WHC-SD-FF-CSER-004, Rev. 1, App. C, pg. 43</td>
</tr>
<tr>
<td>5 Ident-69s mixed with DFAs filling remaining CCC tubes, with DFA in center tube</td>
<td>X</td>
<td>0.9394</td>
<td>WHC-SD-FF-CSER-004, Rev. 1, App. C, pg. 43</td>
</tr>
<tr>
<td>5 Ident-69s mixed with DFAs filling remaining CCC tubes, with DFA in center tube</td>
<td>X</td>
<td>0.9494</td>
<td>Bounded by 7 DFAs with hot-cell rot.$^{(6)}$</td>
</tr>
</tbody>
</table>

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### Table D6-4. Interim Storage Cask $k_{eff}$ Analysis Summary Results. (3 sheets)

<table>
<thead>
<tr>
<th>CCC loading(1)</th>
<th>Hazardous Condition(s) present</th>
<th>Upper bound on $k_{eff}$(2)</th>
<th>Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>&gt;1 Ident-69s mixed with DFAs filling remaining CCC tubes, with Ident-69 in center tube</td>
<td>Misloaded X, Flooded(2) X, Tipped X, Hot-cell rot 1.0139</td>
<td>WHC-SD-FF-CSER-004, Rev. 1, App. C, pg. 42 (up to 3 Ident-69s were found to be acceptable, but were not allowed to simplify loading procedures).</td>
<td></td>
</tr>
<tr>
<td>6 metal fuel assemblies with center tube empty(3)</td>
<td></td>
<td>0.4136</td>
<td>WHC-SD-FF-CSER-004, Rev. 1-B</td>
</tr>
<tr>
<td>6 metal fuel assemblies with center tube empty(3)</td>
<td>X</td>
<td>0.8241</td>
<td>WHC-SD-FF-CSER-004, Rev. 1-B</td>
</tr>
<tr>
<td>6 metal fuel assemblies with center tube empty(3)</td>
<td>X</td>
<td>0.5064</td>
<td>Bounded by hot-cell rot case for 5 metal assemblies with 2 DFAs.</td>
</tr>
<tr>
<td>6 metal fuel assemblies with center tube empty(3)</td>
<td>X</td>
<td>0.8964</td>
<td>WHC-SD-FF-CSER-004, Rev. 1-B (36 in. fuel height caused by tipping, ducts intact).</td>
</tr>
<tr>
<td>&gt;1 metal assembly mixed with DFAs filling remaining CCC tubes, with metal assembly in center tube</td>
<td>Misloaded X, Flooded(2) X, Tipped X, Hot-cell rot 0.9693</td>
<td>WHC-SD-FF-CSER-004, Rev. 1-B (36 in. fuel height caused by tipping, ducts intact).</td>
<td></td>
</tr>
<tr>
<td>&gt;1 metal assembly mixed with DFAs filling remaining CCC tubes, with metal assembly in center tube</td>
<td>Misloaded X, Flooded(2) X, Tipped X, Hot-cell rot 0.8999</td>
<td>WHC-SD-FF-CSER-004, Rev. 1-B</td>
<td></td>
</tr>
<tr>
<td>&gt;1 metal assembly mixed with DFAs filling remaining CCC tubes, with metal assembly in center tube</td>
<td>Misloaded X, Flooded(2) X, Tipped X, Hot-cell rot 0.4682</td>
<td>WHC-SD-FF-CSER-004, Rev. 1-B</td>
<td></td>
</tr>
<tr>
<td>5 metal assemblies mixed with DFAs filling remaining CCC tubes, with DFA in center tube</td>
<td>Misloaded X, Flooded(2) X, Tipped X, Hot-cell rot 0.9293</td>
<td>WHC-SD-FF-CSER-004, Rev. 1-B (36 in. fuel height caused by tipping, ducts intact).</td>
<td></td>
</tr>
</tbody>
</table>
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Table D6-4. Interim Storage Cask $k_{eff}$ Analysis Summary Results. (3 sheets)

<table>
<thead>
<tr>
<th>CCC loading(1)</th>
<th>Hazardous Condition(s) present</th>
<th>Upper bound on $k_{eff}$(3)</th>
<th>Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Misloaded Flooded(2) Tipped Hot-cell rot</td>
<td></td>
<td>Reference</td>
</tr>
<tr>
<td>5 metal assemblies mixed with DFAs filling remaining CCC tubes, with DFA in center tube</td>
<td>X</td>
<td>X</td>
<td>0.7625</td>
</tr>
<tr>
<td>5 metal assemblies mixed with DFAs filling remaining CCC tubes, with DFA in center tube</td>
<td>X</td>
<td></td>
<td>0.5064</td>
</tr>
<tr>
<td>5 metal assemblies mixed with DFAs filling remaining CCC tubes, with DFA in center tube</td>
<td>X</td>
<td></td>
<td>0.8646</td>
</tr>
<tr>
<td>2 metal assemblies mixed with 5 Ident-69s (Ident-69s are not allowed with metal assemblies)</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
</tbody>
</table>

(1) Table D6-4 addresses fully loaded CCCs. The $k_{eff}$ for partially loaded CCCs are bounded by the fully loaded configurations.

(2) The flooded condition for storage of FFTF fuel in the ISC has been determined to be not credible.

(3) This number represents the $k_{eff}$ calculated in the analysis, plus an additional margin to account for bias and uncertainty, as reported in the CSER.

(4) Administrative controls on Ident-69 containers limit the number of pins to 217 (same as DFAs). However, Ident-69 containers could hold up to 233 pins. If all 6 Ident-69 containers are loaded with 233 pins, the total number of pins is still less than an ISC loaded with 7 DFAs (1,398 pins vs. 1,519 pins). Also, the geometry of fuel in an ISC with one tube empty is less optimal for the hot-cell rot hazard than a fully loaded ISC. Therefore, the 7 DFA case bounds the 6 Ident-69 case for hot-cell rot.

(5) Mixtures of DFAs and Ident-69s were required to minimize the number of ISCs.

(6) This case was not analyzed in the CSERs and could contain slightly more fuel than an ISC loaded with 7 DFAs, if the Ident-69 containers are overloaded with 233 pins. However, the Ident-69 pin containers are not expected to be impacted by hot-cell rot. An ISC loaded with DFAs and Ident-69 containers (with failed fuel but intact containers) is judged to be less reactive than an ISC loaded with 7 DFAs subject to complete structural failure (fuel and duct failure) from hot-cell rot.

(7) Only six metal assemblies were made.

CCC = core component container.  
DFA = driver fuel assembly.  
ISC = interim storage cask.
The assumptions used in the analysis are as follows:

- The fuel modeled is outer driver fuel, Type 4.1, with 29.28 wt% Pu (239 and 241) and 11.63 wt% Pu-240, which is the highest total Pu enrichment for any of the FFTF fuel pins.

- Unirradiated fuel was modeled; no credit was taken for burnup.

- The radial dividers in the Ident-69 were neglected (resulting in a positive bias in $k_{\text{eff}}$).

The model was validated against experiments with fast test reactor fuel pins in water, as reported in "Critical Experiments with Fast Test Reactor Fuel Pins in Water" (Bierman et al. 1979). The validation showed a small positive bias in $b_N$ of ~ 0.006 for the MCNP results when compared to the benchmark experiments. In addition, omitting the radial dividers from the Ident-69 model introduced another positive bias of 0.013 in $k_{\text{eff}}$.

**Criticality Evaluation for the Long-Term Storage of Fast Flux Test Facility Fuel in Interim Storage Containers (Rev. 0)**

WHC-SD-FF-CSER-004 (Rev. 0), Criticality Evaluation for the Long Term Storage of FFTF Fuel in Interim Storage Containers, analyzed the $k_{\text{eff}}$ of ISCs loaded with seven DFAs subject to hot-cell rot. The Monte Carlo criticality computer code, MONK6B, was used for the calculations of $k_{\text{eff}}$. Various geometries resulting from disintegration of the fuel due to hot-cell rot were studied, as follows:

- A set of parametric cases was analyzed that assumed complete fuel and cladding failure with intact assembly ducts. In this model, fuel and cladding were uniformly mixed within the intact duct. Heights from 21 in. to 36 in. for the uniform mixture were analyzed. Both dry and flooded configurations were analyzed.

- A case similar to the above cases but assuming failure of the assembly duct was also analyzed. The homogeneous mixture of fuel, cladding, and duct debris was found to have a minimum height of 12 in. Both dry and flooded configurations were analyzed. Fuel debris in pellet or powder form was also analyzed for the flooded configuration.

- A dry case was analyzed assuming that the cladding and duct debris were separated from the fuel debris, resulting in a minimum 7-in. fuel height.

The geometries analyzed provide an upper bound on $k_{\text{eff}}$ for the possible configurations resulting from hot-cell rot in an ISC loaded with DFAs.
The assumptions used in the analysis are as follows:

- The fuel modeled is outer driver fuel, Type 4.1, with 29.28 wt% Pu (239 and 241) and 11.63 wt% Pu-240, which is the highest total Pu enrichment for any of the FFTF fuel pins.

- Unirradiated fuel was modeled; no credit was taken for burnup.

The analysis also considered storage of multiple ISCs on a pad by analyzing an infinite planar array of adjacent, flooded ISCs. However, large scale flooding of the ISA is not considered credible in the hazard analysis discussed in Section D6.3.2.

In addition, the possibility of fuel heights larger than the nominal 36 in. was considered. These larger fuel heights were postulated as the result of fuel and cladding debris spreading out within the assembly duct in a horizontal ISC (e.g., due to tipping). Assembly ducts were assumed to be intact for the extended height cases.

Validation of the MONK6A computer code for WHC-SD-FF-CSER-003, *Criticality Evaluation for the Long Term Storage of FFTF Fuel in Interim Storage Containers*, can be applied to the MONK6B computer code. WHC-SD-SQA-CSWD-20019, *MONK6B Pu Validation*, demonstrated the equivalence of MONK6B to MONK6A, as the cross-section library of the two versions is identical. To validate the accuracy of the MONK6A model, WHC-SD-FF-CSER-003 made comparison calculations with the previous criticality analyses for intact DFAs in WHC-SD-FF-CSER-002, *Criticality Safety Evaluation Report of Fuel Assemblies in Core Component Containers*. The only difference between the calculations was that the MCNP calculations in WHC-SD-FF-CSER-002 used explicitly modeled fuel pins, while the MONK6A calculations used homogenized atom densities for the fuel pins and water. The MONK6A model was found to give a slightly larger $k_{eff}$ that was within the statistical uncertainty of the calculation at the 95% confidence level. The validation document, WHC-SD-SQA-CSWD-20015, *MONK6A Pu Validation*, states that a calculated value of $k_{eff} < 0.935$ is required to assure $k_{eff} < 0.95$, including bias and uncertainty. Direct comparisons of MONK6A calculations against critical experiments show that calculated $k_{eff}$ values less than 0.935 ensure that $k_{eff} < 0.95$, including uncertainty and bias at the 95% confidence level. For comparison to the safe subcritical limit of 0.95, a value of 0.015 was added to the calculated values of $k_{eff}$ to obtain the values in Table D6-4.

**Criticality Evaluation for the Long-Term Storage of Fast Flux Test Facility Fuel in Interim Storage Containers (Rev. 1)**

WHC-SD-FF-CSER-004 (Rev. 1) adds Appendix C to Rev. 0. Appendix C analyzes mixtures of DFAs and Ident-69s for flooded conditions. This analysis was performed to allow combinations of DFAs and Ident-69 containers to minimize the number of ISCs required in the storage area. The mixed configurations were analyzed using the same fuel assumptions.
discussed previously for the DFAs. All pin containers were assumed to be optimally loaded with 97 pins. The following cases were analyzed:

- One to seven Ident-69s with the remaining tubes occupied by DFAs. The center CCC tube is occupied by an Ident-69.
- Zero to five Ident-69s with one outer tube empty and the center tube with a DFA.
- Six Ident-69s in the outer tubes.
- Five Ident-69s in the outer tubes and two DFAs in the remaining tubes.

The analysis also considered storage of multiple ISCs on a pad, with mixtures of Ident-69s and DFAs, by analyzing an infinite planar array of adjacent, flooded ISCs with five Ident-69s and two DFAs. However, large scale flooding of the ISA is not considered credible in the hazard analysis discussed in Section D6.3.2.

**Criticality Evaluation for the Long-Term Storage of Fast Flux Test Facility Fuel in Interim Storage Containers (Rev. 1-A)**

WHC-SD-FF-CSER-004 (Rev. 1-A) is written as Appendix D to WHC-SD-FF-CSER-004. This document contains analysis of the k_{eff} of modified Ident-69s containing returned FFTF experimental fuel and repackaged fuel in a spent fuel container, as described in Section D6.4.2. The experimental fuel is from the following FFTF test assemblies: ACO-3, ACO-1, FO-2, MFA-1, AAD-5, DE4-1, D9-1, DE2-2, D9-4 and C-1. The pins from each of these assemblies have previously been classified as equivalent to MOX fuel for the purposes of criticality safety. The MONK6B computer code was used to calculate k_{eff}. The code validation discussion for WHC-SD-FF-CSER-004 (Rev. 1) is applicable to Appendix D as well. The configurations of the flooded, water-reflected, modified Ident-69 analyzed are as follows:

- 48 stainless steel tubes inside a spent fuel canister with 4 kg of MOX fuel homogeneously distributed within each tube volume. Fuel within the tubes is dry, all other void spaces are flooded. Tube pitch within the spent fuel canister was varied from 0.50 in. (close packed) to 0.67 in. (spread out to the volume of Ident-69).
- 4 kg of MOX fuel in pellet form (.494 cm diameter spheres) inside a flooded spent fuel canister. Pellet pitch was varied from 0.20 in. (close packed) to 0.45 in. (homogeneous distribution throughout spent fuel canister). The stainless steel of the fuel tubes was neglected.
- 4 kg of MOX fuel in pellet form (0.494 cm diameter spheres) inside a flooded, modified Ident-69. Pellet pitch was varied from 0.20 in. (close packed) to 0.88 in. (homogeneous distribution inside the modified Ident-69). The stainless steel of the spent fuel canister and the fuel tubes was neglected.
4 kg of MOX fuel in powder form inside a flooded, modified Ident-69. The stainless steel of the spent fuel canister and the fuel tubes was neglected.

The calculated $k_{\text{eff}}$ values for these configurations were compared to a calculated $k_{\text{eff}}$ value for a flooded, water-reflected, optimized Ident-69 loaded with 97 pins, as analyzed in WHC-SD-FF-CSER-004 (Rev. 1).

The following assumptions were made in the analysis:

- The modified Ident-69s contain $<4$ kg of experimental pins and repackaged fuel.
- Fuel pin cladding was ignored.

Criticality Evaluation for the Long-Term Storage of Fast Flux Test Facility Fuel in Interim Storage Containers (Rev. 1-B)

WHC-SD-FF-CSER-004 (Rev. 1-B) presents analysis of the $k_{\text{eff}}$ of CCCs loaded with mixtures of metal fuel assemblies and DFAs. The MONK6B computer code was used to determine $k_{\text{eff}}$ values. The following cases were analyzed:

- One to seven metal fuel assemblies in a CCC with a metal assembly in the center tube and the remaining tubes occupied by DFAs. The ISC was assumed to be flooded. Fuel and cladding were assumed to be homogeneously distributed over a 36-in. height (nominal fuel height) within intact fuel assembly ducts.

- Four or five metal assemblies with a DFA in the center tube and the remaining tubes occupied by DFAs. Dry and flooded configurations were analyzed. Fuel and cladding were assumed to be homogeneously distributed over a 36-in. height (nominal fuel height) or over a 12 in. height (minimum height of collapsed fuel and cladding debris) within intact fuel assembly ducts. Dry and flooded configurations were analyzed. Several cases with dry and flooded intact fuel were also considered.

- Five metal assemblies with two DFAs (one in the center tube) within a flooded ISC. Fuel and cladding were assumed to be homogeneously distributed over 36-in. to 140-in. heights (nominal fuel height) within intact fuel assembly ducts. This case represents a horizontal ISC (e.g., due to tipping during movement or vehicle collision).

- Six metal assemblies with the center tube occupied or unoccupied. The ISC was assumed to be flooded. Fuel and cladding were assumed to be either intact or completely failed due to hot-cell rot. For the hot-cell rot cases, the fuel and cladding were homogeneously distributed over a 36-in. height (nominal fuel height) within intact fuel assembly ducts.
Seven metal assemblies. The ISC was assumed to be flooded. Fuel and cladding were assumed to be homogeneously distributed over a 36-in. height (nominal fuel height) within intact fuel assembly ducts.

Several configurations having mixtures of metal assemblies and Ident-69s were also analyzed, which assumed intact ducts, a flooded ISC, and the fuel and cladding homogeneously distributed over a 36-in. height due to hot-cell rot.

The analysis used the following assumptions:

- Metal fuel has 32.4 wt% U-235 enrichment.
- Unirradiated fuel was used; no credit was taken for burnup.

Validation of the MONK6B computer code is discussed previously in the summary of WHC-SD-FF-CSER-004 (Rev. 1). For comparison to the safe subcritical limit of 0.95, a value of 0.015 for bias plus uncertainty was added to the calculated values of $k_{eff}$ and reported in Table D6-4.

**D6.3.3.1.2 Analysis Results.** Analysis results are summarized in Table D6-4. The table contains entries for all fully loaded CCC configurations and all credible hazardous conditions identified in the hazard evaluation. In addition, some CCC configurations that were found to be not credible are also presented, since they were analyzed in the CSERs. Some of these cases have $k_{eff}$ values beyond the acceptable limit, but they do not require administrative or engineered controls since they result from hazardous conditions that were considered incredible in the hazard evaluation (Section D6.3.2).

**Normal Conditions**

Normal conditions consist of dry ISCs loaded with structurally intact DFAs, Ident-69s, metal assemblies, or modified Ident-69s subject to the restrictions described in Section D6.3.1.

From Table D6-4, the highest $k_{eff}$ for normal configurations in the table (i.e., no hazardous conditions present) was calculated to be 0.4125, including bias and uncertainty, for an ISC loaded with six metal assemblies. Metal assemblies were found to be slightly more reactive than DFAs. This upper limit for normal configurations is well below the safe subcritical limit of 0.95.

For a dry ISC loaded with intact fuel, Ident-69s were found to be less reactive than DFAs. The $k_{eff}$ for intact dry fuel increases with decreasing pin pitch. The pin pitch for the DFA is 0.284 in., while the pin pitch for 217 pins within an Ident-69 is greater than 0.30 in. (WHC-SD-FV792-DA-004). Thus, the most reactive dry configuration for Ident-69s with 217 intact pins is less reactive than a dry DFA with 217 pins.
Results of the analysis for the storage of multiple ISCs indicate that the CCCs are neutronically separated within the ISC. Compared to the $k_{\text{eff}}$ of a single ISC, model results for an infinite planar array of adjacent ISCs showed an increase in $k_{\text{eff}}$ on the order of the statistical uncertainty in $k_{\text{eff}}$. Thus, for arrays of dry ISCs with intact fuel, $k_{\text{eff}}$ for the array will still fall well below the safe subcritical limit of 0.95.

Abnormal Conditions

As discussed in Section D6.3.2, individual hazardous events that were determined to be credible, but unlikely, are as follows:

- Misloading of a cask prior to transport to the ISA.
- Tipping or dropping of a cask during handling (structural damage to the CCC due to a design basis drop was shown to be not credible in Chapter D3.0).

The results of the analysis in WHC-SD-FF-CSER-004 (Rev. 1-A) for experimental fuel in a modified Ident-69 container show that for all credible hazardous conditions, the $k_{\text{eff}}$ for the modified Ident-69 will be bounded by $k_{\text{eff}}$ for an optimized, unmodified Ident-69 container, as analyzed in WHC-SD-FF-CSER-004 (Rev. 1). Therefore, all hazardous ISC configurations with modified Ident-69 containers are bounded by the configuration obtained when the modified Ident-69 containers are replaced by optimized unmodified Ident-69 containers.

The hazard analysis also identified hot-cell rot as an anticipated hazard over the 40-year storage lifetime of the casks. The actual likelihood of hot-cell rot affecting the fuel pins is highly uncertain but has been observed at several other facilities. The thickness of the stainless steel in the fuel assembly ducts is several times greater than the fuel cladding itself and is less susceptible to corrosion than the fuel cladding. Ident-69 containers have not been exposed to thermal cycling and the corrosive environment of the sodium coolant during the cleaning process prior to loading in the ISCs and are not expected to suffer from the effects of hot-cell rot. However, in order to provide an upper bound on the likelihood of the hazard, the hazard analysis assumed that the event will occur and that fuel pins and/or assembly ducts may suffer complete structural failure. The CCC tubes were assumed to be unaffected by any corrosion mechanisms due to their robust design requirements. Hot-cell rot was modeled in the $k_{\text{eff}}$ analyses by either (1) assuming that fuel and cladding was homogeneously mixed within the void space of the ducts, or (2) assuming that fuel, cladding, and assembly duct material were homogeneously mixed within the void space of the CCC tube. The height of the fuel and cladding debris, or fuel and cladding and duct debris, was varied parametrically in the $k_{\text{eff}}$ calculations to determine the geometric configuration with the largest $k_{\text{eff}}$. 

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Annex D - 200 Area Interim Storage Area
**k\textsubscript{eff} of Configurations Due to Hot-cell Rot**

The largest value of \(k_{\text{eff}}\) (including bias and uncertainty at the 95% confidence level) for a dry, normally loaded ISC that has experienced hot-cell rot was found to be 0.9494. This extreme case of hot-cell rot was obtained for a CCC loaded with seven DFAs by assuming the entire assembly has disintegrated. In addition, the stainless steel of the cladding and ducts was assumed to have separated from the fuel debris by some unknown mechanism. The separated fuel was compacted to its minimum possible height of 7.2 in. within the bottom of the CCC tubes to form the most reactive configuration constrained only by the CCC tube geometry.

**k\textsubscript{eff} of Single Contingency Configurations**

Single-contingency configurations are defined as those that occur due to a single unlikely hazardous event (with or without hot-cell rot), since some form of hot-cell rot was assumed to occur in the hazard analysis. Single-contingency events were assumed to include the possibility of hot-cell rot combined with one other unlikely hazardous event.

**Misloaded cask** – From Table D6-4, the most reactive misloaded cask configuration with intact fuel pins was found to be five metal fuel assemblies and two DFAs in the CCC, with a metal assembly in the center position (a configuration with six metal assemblies and one DFA was not analyzed, but should give a slightly higher \(k_{\text{eff}}\) — still far below 0.95). Metal assemblies are slightly more reactive than DFAs.

Note: Misloading of a cask with seven Ident-69 containers was considered to be not credible by the hazard evaluation and was not systematically analyzed in the CSERs. However, two cases with seven Ident-69 containers were analyzed and are reported in Table D6-4 for information only.

**Tipped cask** – Tipping or dropping a cask could result in damaged fuel that is spread out within the void space of the CCC tubes. The \(k_{\text{eff}}\) of a tipped cask with dry fuel is bounded by the analysis for hot-cell rot (discussed previously), since dry fuel has been shown to be more reactive when placed in the most compact geometry. Tipped or dropped casks with dry fuel were not analyzed in the CSERs and are not shown in Table D6-4. Tipping or dropping a flooded cask was analyzed in the CSERs and is discussed below under "Flooded Cask."

**k\textsubscript{eff} of Other Hazardous Conditions**

The remaining hazardous conditions that have not been excluded as not credible by the hazard evaluation are as follows:

**Misloaded and Tipped Cask** – This configuration could result in damaged fuel spread out within the void space of the CCC tubes. The \(k_{\text{eff}}\) for the system is bounded by the misloaded configuration with hot-cell rot (discussed previously) since dry fuel was shown to be more reactive when compacted to its minimum possible height.
Incredible Hazardous Conditions

Hazardous conditions found to be not credible by the hazard evaluation, but were analyzed in the CSERs are as follows:

Flooded Cask – Cask flooding at the ISA was judged to be not credible by the hazard evaluation. However, flooded configurations analyzed in the CSERs are reported in Table D6-4 for completeness. The flooded configurations were analyzed in the CSERs to ensure that fuel remained safely subcritical during fuel handling operations, including washing activities.

From Table D6-4, the most reactive flooded, normally loaded cask configuration with intact fuel pins was found to be five metal fuel assemblies and two DFAs in the CCC, with a DFA in the center position. This configuration resulted in an upper bound on $k_{\text{eff}}$ of 0.9380, including bias and uncertainty at the 95% confidence level. Including the possibility of hot-cell rot raises the $k_{\text{eff}}$ slightly to 0.9453. The hot-cell rot configuration assumed that the fuel/cladding water debris extended over the normal height of the fueled region (36 in.) in order to achieve optimal moderation. In reality, hot-cell rot would tend to create debris that collects at the bottom of the CCC tube, which is a less optimally moderated geometry.

The infinite flooded array of ISCs containing five Ident-69s and two DFAs with intact fuel pins was calculated to have a $k_{\text{eff}}$ near the safe subcritical limit in WHC-SD-FF-CSER-004 (Rev. 1). However, large scale flooding of the ISA is not considered credible in the hazard analysis discussed in Section D6.3.2.

Other flooded configurations were analyzed in the CSERs. Some of these configurations resulted in $k_{\text{eff}}$ greater than the safe subcritical limit of 0.95. The highest $k_{\text{eff}}$ found in the analysis occurred for a tipped, flooded ISC loaded with seven DFAs that have experienced complete failure due to hot-cell rot. The $k_{\text{eff}}$ for this configuration was found to be 1.13. Additional criticality controls for flooded configurations for which $k_{\text{eff}} > 0.95$ are not necessary, as these conditions are not considered credible by the hazard evaluation.

D6.3.3.2 NRF TRIGA Fuel.

**D6.3.3.2.1 Analysis Models.** The $k_{\text{eff}}$ of normal and accident conditions for TRIGA fuel stored in the NRF TRIGA casks has been analyzed in the CSERs prepared for the handling and storage of TRIGA fuel (WHC-SD-TP-SARP-008, WHC-SD-FF-CSER-006, and WHC-SD-SQA-CSA-30006). The models used in these documents are summarized below.

Safety Analysis Report for Packaging (Onsite) NRF TRIGA Packaging

WHC-SD-TP-SARP-008 contains the analysis of infinite planar arrays of TRIGA casks to show the casks meet the criticality requirements for transportation from the 300 Area to the 400 Area ISA at FFTF. As such, the explicit geometry and material of the Rad-Vault are not
modeled. However, the infinite planar array of close-packed casks analyzed can be taken to represent an upper bound on casks in the Rad-Vault, which will be in a less reactive geometry.

The calculational model for MCNP included all components of a TRIGA cask loaded with 18 fuel elements. The cladding, fuel basket, inner container, and cask inner and outer shell with lead shielding were explicitly modeled. Normal fuel geometries inside dry cask arrays were analyzed with and without water occupying the interstitial space between the casks. The normal fuel geometry cases with no water occupying the space between casks represents an upper bound on the Rad-Vault configuration, since there will only be six fully fueled casks in the Rad-Vault, and the cask space will be larger than the arrays analyzed here.

In addition, abnormal fuel conditions were analyzed by removing the fuel cladding and stainless steel fuel baskets from the model. The fuel pitch inside the inner container was then varied to determine the worst-case geometry for the abnormal configuration (the geometry of the fuel elements themselves remained intact). The abnormal configurations were analyzed with or without water inside the casks or occupying the interstitial area between the infinite planar array of casks. A single case was analyzed with a homogeneous distribution of fuel and water within the region of the inner container.

The MCNP criticality computer code was used to perform the $k_{eff}$ analysis for the infinite cask arrays. Uncertainty in the $k_{eff}$ values for normal fuel geometry cases was estimated by combining the statistical uncertainty reported by MCNP with rough estimates for the error in cross-section, physical modeling, and dimensional tolerances. For normal configurations, this uncertainty in $k_{eff}$ was determined to be 0.0036, which represents the statistical uncertainty in the Monte Carlo analysis (Breimeister 1993). For the abnormal cases, a configuration error was combined with the other error factors to account for the possibility that the limited parametric study performed on the fuel geometry and water density variations did not obtain the worst-case configuration. The overall uncertainty for abnormal cases was determined to be 0.0057. MCNP has been validated against experimental benchmarks by the code development group at Los Alamos National Laboratory (Whalen. et al. 1991). The validation experiments included critical assemblies for low-enriched uranium systems, graphite and water-reflected systems, fast neutron systems (Godiva and Jezebel assemblies), and interactive (array) units. The code showed agreement to the experimental $k_{eff}$ values within 1% for all benchmarks.

Criticality Safety Evaluation Report for TRIGA Fuel Storage at 400 Area Interim Storage Area

WHC-SD-FF-CSER-006, Criticality Safety Evaluation Report for TRIGA Fuel Storage at 400 Area Interim Storage Area, provides analysis of the abnormal fuel configuration resulting from the fall of a large crane onto the Rad-Vault.

The TRIGA cask safety analysis report for packaging (WHC-SD-TP-SARP-008) determined that the cask structural integrity would be maintained in the worst-credible drop or vehicle collision accident. However, it was postulated that a large crane falling onto the
Rad-Vault could result in catastrophic structural failure of the vault and TRIGA casks, rearranging the fuel into a more reactive geometry than normal. With no geometry control, there is sufficient fuel in the Rad-Vault to go critical under idealized conditions.

The neutron transport code, WIMS-E, was used to parametrically vary the spacing of both the aluminum-clad and the stainless steel-clad TRIGA fuel elements in water. These heterogeneous lattice calculations were performed using the British Atomic Energy Authority 69-group nuclear cross-section library. Once the transport calculations were complete, WIMS-E used the 69-group radial flux solution to collapse cross-sections to a two-group set for the smeared lattice cell. The GOLF computer code then used these two-group cross-sections and performed neutron diffusion calculations to find the finite spherical and hemispherical radii required for $k_{eff} = 0.95$, given full water reflection.

The GOLF code has been validated for calculating critical dimensions for idealized geometries (such as spherical and hemispherical), when using nuclear cross-sections obtained from the WIMS-E lattice code. These validations are described in various sources, such as for uranium metal billets, N Reactor Mark IA critical experiments, lattices of uranium metal rods in water with various enrichments and rod sizes (including boron poisoned water lattices), and uranium and plutonium nitrate solutions. For the Mark IA critical experiments, WIMS-E and GOLF were used to predict the critical radius of the two-tier cylindrical arrangement that was measured to be critical. The critical prediction was slightly conservative (i.e., the predicted cylinder radius was less than what was measured as critical). The MCNP computer code was used to predict the $k_{eff}$ of the WIMS-E/GOLF prediction ($k_{eff} = 0.99203 \pm 0.0017$). This establishes a WIMS-E/GOLF bias of 0.008, which was added to the calculated results for $k_{eff}$ and then reported in Table D6-5. Statistical uncertainties in the WIMS-E/GOLF results were not reported in the analysis. Independent verification has been performed for WIMS-E/GOLF critical mass predictions using both the MCNP and MONK5 Monte Carlo computer codes (Schwinkendorf 1990; WHC-SD-NR-CSER-007, Appendices A and B; WHC-IP-0840-FMEF).

The analysis assumed a worst-case random geometry of fuel occurred as a result of the accident. The most reactive geometry that could credibly occur is a hemispherical pile of homogeneously distributed fuel and cladding. This geometry was analyzed with and without material from the casks and fuel baskets. Varying degrees of moderation in the rubble pile were also analyzed.

Criticality Safety Evaluation Report 95-012: Transfer of TRIGA Fuel from 308 Building Reactor Pool to Storage Casks

WHC-SD-SQA-CSA-30006, Criticality Safety Evaluation Report 95-012: Transfer of TRIGA Fuel from 308 Building Reactor Pool to Storage Casks, contains analysis of the $k_{eff}$ of normal and hazardous conditions that may be encountered during transfer operations from the TRIGA reactor pool into TRIGA casks. The analysis is directed at the transfer operations and does not explicitly consider hazards associated with long-term storage of the fuel. Geometric configurations with the Rad-Vault are not directly analyzed. However, the case analyzed for a
close-packed array of seven normally loaded casks can be taken as an upper bound on the $k_{\text{eff}}$ of the actual Rad-Vault configuration. The cask array was modeled as a normally configured, dry cask with 18 aluminum clad or stainless steel clad fuel elements inserted in the aluminum fuel basket.

The MONK6B computer code was used to perform the $k_{\text{eff}}$ analysis for the cask array. The documentation for the code includes results of validation calculations for several critical uranium-enriched systems. The benchmark experiments were performed on highly enriched systems or low-enriched systems, as opposed to the 20% enriched TRIGA fuel. WHC-SD-SQA-CSA-30006 judged the validity of the code for TRIGA fuel based on the benchmark results for highly and low-enriched critical experiments. The results of the validation calculations show a consistent positive bias in $k_{\text{eff}}$ for the MONK6B calculations. This positive bias was not included in the reported values for $k_{\text{eff}}$ in Table D6-5. Statistical uncertainties at the 2σ level were added to the calculated values of $k_{\text{eff}}$ and the result was reported in Table D6-5.

Table D6-5. NRF TRIGA Cask $k_{\text{eff}}$ Analysis Summary Results.

<table>
<thead>
<tr>
<th>TRIGA cask loading</th>
<th>Hazardous Condition(s) Present</th>
<th>Upper Bound on $k_{\text{eff}}$</th>
<th>Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>18 SS fuel pins</td>
<td>Flooded</td>
<td>Tipped</td>
<td>Crushed</td>
</tr>
<tr>
<td>18 AL fuel pins</td>
<td>X</td>
<td></td>
<td></td>
</tr>
<tr>
<td>18 SS fuel pins</td>
<td>X</td>
<td></td>
<td></td>
</tr>
<tr>
<td>18 SS fuel pins</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>18 SS fuel pins</td>
<td>X</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>18 AL fuel pins</td>
<td>X</td>
<td></td>
<td></td>
</tr>
<tr>
<td>18 AL fuel pins</td>
<td>X</td>
<td></td>
<td></td>
</tr>
<tr>
<td>18 SS fuel pins</td>
<td>X</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

AL = aluminum.
SS = stainless steel.
D6.3.3.2.2 Analysis Results. Analysis results for $k_{eff}$ of the TRIGA casks are summarized in Table D6-5. The table contains entries for all cask configurations and all combinations of hazardous conditions identified as credible in the hazard evaluation. In addition, some cask configurations that were considered not credible by the hazard evaluation are also presented for information, since they were analyzed in the CSERs.

Normal Conditions

Normal conditions for the TRIGA casks consist of six TRIGA casks loaded with up to 18 intact dry fuel assemblies. Normal configuration analyses were performed for arrays of close-packed casks or an infinite lattice of close-packed casks. This geometry provides an upper bound on $k_{eff}$, since the actual geometry in the Rad-Vault includes extra spacing and absorber material due to the DOT-6M casks and the empty 55-gal. drum inside the vault. Inclusion of the DOT-6M casks would lower the $k_{eff}$ compared to the analyzed geometries.

Analysis in WHC-SD-SQA-CSA-30006 found that the upper bound on $k_{eff}$ for an array of seven normally loaded casks was 0.5818. The analysis shows that all normal conditions for casks loaded in the Rad-Vault have an upper bound on $k_{eff}$ (including bias and uncertainty at the 95% confidence level), which is well below the safe subcritical limit of 0.95.

Abnormal Conditions

Hazardous conditions resulting from a single credible but unlikely hazardous event for the TRIGA fuel are represented by the following:

- Flooded configurations.
- Tipped/dropped configurations with fuel rubble in abnormal geometries due to fuel pin failure.
- Crushed configurations with fuel and/or cask rubble in abnormal geometries due to catastrophic failure of the Rad-Vault and casks.

Various combinations of these hazardous events were analyzed.

Note: Misloaded casks are not possible as there are no loading restrictions on the type of fuel elements loaded into the casks, and the fuel basket is to be fully loaded with the maximum number of elements.

Single-Contingency Hazardous Conditions

Single-contingency configurations are defined as those due to a single unlikely hazardous event.
Flooded Casks – The worst-case flooded cask condition analyzed in the CSERs was found to have an upper bound on $k_{\text{eff}}$ of 0.7564, including bias and uncertainty at the 95% confidence level. The configuration analyzed in WHC-SD-SQA-CSA-30006 was a seven-cask array of fully loaded casks surrounded by water. Only the central cask in the array was flooded, with the outer six casks dry inside, which does not represent the worst-case flooding configuration. However, $k_{\text{eff}}$ for a configuration with all seven casks flooded is bounded by the flooded and tipped configuration discussed below, which assumed an infinite planar array of casks and neglected fuel cladding and fuel basket stainless steel. The $k_{\text{eff}}$ for the worst-case tipped and flooded hazardous configuration was found to fall below the safely subcritical limit, and, therefore, all configurations with flooding alone will be safely subcritical.

Tipped/Dropped Casks – The worst-case cask configuration resulting from fuel damage due to cask tipping or dropping was analyzed in WHC-SD-TP-SARP-008. The configuration analyzed was an infinite planar array of dry casks fully loaded with fuel pins. Cladding and fuel basket materials were neglected to represent the effects of fuel damage. The upper bound on $k_{\text{eff}}$ for this configuration was found to be 0.799, which falls below the safe subcritical limit of 0.95.

Crushed Casks – The worst-case crushed cask configuration found in Table D6-5 gave a $k_{\text{eff}}$ of 0.858. The configuration analyzed in WHC-SD-FF-CSER-006 was an infinite planar array of optimally moderated, close-packed, intact fuel rods. The stainless steel of the inner canister was co-mingled with the fuel. This infinite planar array configuration provides an upper bound on the $k_{\text{eff}}$ of actual configurations that might result from the crushing of the Rad-Vault and dispersal of its contents. Separation of the inner canister stainless steel was not considered credible, but was analyzed in the CSER and is discussed below. This idealized debris configuration indicates that actual configurations resulting from crushing of the Rad-Vault will have a $k_{\text{eff}}$ below the safe subcritical limit of 0.95.

Other Credible Hazardous Conditions

Tipped/Dropped and Flooded Casks – The worst-case cask configuration resulting from a flooded Rad-Vault and casks with fuel damage due to tipping or dropping was analyzed in WHC-SD-TP-SARP-008. The configuration analyzed was a flooded infinite planar array of internally flooded casks fully loaded with fuel pins. Cladding and fuel basket materials were neglected to represent the effects of fuel damage. The upper bound on $k_{\text{eff}}$ for this configuration was found to be 0.939, which falls below the safe subcritical limit of 0.95.

Crushed and Flooded Casks – The hazardous configuration resulting from a combination of crushed and flooded casks is bounded by the analysis for crushed casks (discussed previously) since that analysis already assumed optimal moderation in the debris pile.
Incredible Hazardous Conditions

The most reactive abnormal condition analyzed was a hemispherical pile of fuel rubble resulting from a crane falling onto the Rad-Vault. The results of the analysis in WHC-SD-FF-CSER-006 showed that it is theoretically possible to exceed $k_{\text{eff}} = 0.95$ under optimal conditions of geometry and moderation within the rubble pile. This assumes that all absorbing material in the cask and Rad-Vault debris is somehow preferentially separated from the fuel—an event that is considered extremely unlikely. WHC-SD-FF-CSER-006 also showed that if the steel inner-liner in the cask is included in the rubble pile, the $k_{\text{eff}}$ for an infinite planar array of TRIGA fuel is $< 0.85$. The worst-case scenario consisting of a crane failure, an optimal geometry of the fuel rubble, and separation of absorbing material from the cask is not considered credible. Further engineered or administrative controls to reduce the likelihood of this criticality event are not required.

D6.3.3.3 Commercial Light Water Reactor Fuel.

D6.3.3.3.1 Analysis Models. The $k_{\text{eff}}$ of normal and accident conditions for storage of commercial LWR fuel in NAC-1 casks has been analyzed in the CSERs prepared for the handling and storage of LWR spent fuel (SNF-4875, HNF-4832). The models used in these documents are summarized in the subsections that follow.

Criticality Evaluation for Long-Term Storage of Light Water Reactor Fuel in NAC-1 Casks

SNF-4875, Criticality Evaluation for Long Term Storage of Light Water Reactor Fuel in NAC-1 Casks, contains analysis of the $k_{\text{eff}}$ of NAC-1 casks loaded with PWR assemblies. Most of the materials and geometry of the NAC-1 casks loaded with PWR fuel assemblies from Calvert Cliffs and Point Beach were explicitly modeled in the analysis. The model included the cask inner liner, lead shielding, outer liner, steel lid and cask bottom, and neutron shield tanks. The cask lid and bottom thickness differed slightly from the as-built configuration. This dimension will not affect the $k_{\text{eff}}$ calculations, since the cask is nearly an infinite system in the axial direction, and end effects are negligible. The inner canister was modeled with the design thickness, but the steel in the inner canister ribs and the structural supports for the fuel assembly were neglected. The model did not take credit for the extra steel and spacing between casks provided by the International Standards Organization (ISO) containers, in which each cask will be stored. The fuel was modeled with the following input assumptions:

- Unirradiated fuel was modeled; no credit was taken for burnup.
- The zircaloy cladding was neglected.
- Enrichment for all pins in the model was taken as the highest actual value for any single pin in the assembly.
The casks containing the Point Beach assemblies have a maximum of 179 pins, with 3.2 wt% enrichment.

The cask containing the Calvert Cliffs full assembly has a maximum of 176 pins, with 2.72 wt% enrichment.

The cask containing the Calvert Cliffs partial assembly has a maximum of 144 pins, with 2.73 wt% enrichment.

To determine an upper bound on six normal cask configurations, an infinite planar array of close-packed casks with dry inner containers and undamaged fuel was calculated using KENO-Va. Moderation in the neutron shield tanks and in the interstitial region between the casks was varied using XSDRN, with an infinite 3-dimensional array, to determine the most reactive configuration. This configuration was then analyzed with KENO-Va to determine the upper bound on $k_{\text{eff}}$ for the planar array.

The accident condition analyzed for the casks was a flooding scenario. As discussed in Section D6.3.2, hot-cell rot and fuel rearrangement due to tipping or collisions was not considered credible for the NAC-I casks. Also, vertically stacked arrays were not considered credible. To determine the most reactive flooded configuration, the XSDRN code was used to analyze a single cask flooded internally and surrounded by water reflection. The number of pins in the cask was varied from the maximum of 196 to determine the optimal number. For a given number of pins, the pin pitch was chosen such that the square array filled the entire assembly cross-section. Once the optimal pin number was determined, the KENO-Va code was used to calculate the upper bound on $k_{\text{eff}}$ (including bias and uncertainty) of the most reactive configuration. KENO-Va and not XSDRN was used for the final reported $k_{\text{eff}}$ since KENO-Va has been benchmarked to critical experiments, as discussed below. The optimal configuration for each fuel type was then used to analyze an infinite 3-dimensional array of close-packed casks with XSDRN, as for the dry case. Water density between the casks was varied to determine the most reactive configuration. The most reactive configuration was then analyzed with KENO-Va to determine the upper bound on $k_{\text{eff}}$.

The SCALE 4.2 system of codes was used to model the fuel assemblies in the casks. The NITAWL and XSDRN codes were used to generate a cell-weighted cross-section, and the KENO-Va code was used to model the cask geometry and predict $k_{\text{eff}}$. XSDRN was also used to perform scoping calculations to determine the optimal geometric configuration of the fuel in the flooded cask arrays to determine the optimal moderation between casks in the dry array calculations. XSDRN and not KENO-Va was used for the parametric variations since it is a deterministic code that does not exhibit random variations in $k_{\text{eff}}$, which could mask small changes in $k_{\text{eff}}$ over the parametric variations.
Uncertainty due to dimensional tolerances and statistical uncertainty were included in the final $k_{\text{eff}}$ values reported in Table D6-6 for the KENO-Va calculations. Dimensional uncertainties were included using a one-sided tolerance limit factor, as outlined in "A Systematic Approach to Establishing Criticality Biases" (Larson 1995). For example, the uncertainty in the fuel pellet enrichment, density, and diameter was added to the values reported in Chapter D2.0, which represents an increase in the amount of fissile material to account for uncertainty in input parameters for the fuel. In the single cask calculations, the neutron reflection from the cask structure was increased by adding the dimensional tolerances to the thickness of the lead shielding. On the other hand, the infinite planar array calculations subtracted the dimensional tolerances from the cask dimensions to maximize interaction between the casks.

Code bias in the KENO-Va calculations were also included in the $k_{\text{eff}}$ values reported in Table D6-6 for the flooded arrays. Bias was not included in the dry array calculations. Since the $k_{\text{eff}}$ values are so far below critical for the dry arrays, extrapolation of code bias from critical benchmark experiments would not be justified and is not necessary to ensure that $k_{\text{eff}} < 0.95$.

Table D6-6. NAC-1 Cask $k_{\text{eff}}$ Analysis Summary Results.

<table>
<thead>
<tr>
<th>NAC-1 cask loading</th>
<th>Hazardous condition(s) present</th>
<th>Upper bound on $k_{\text{eff}}$</th>
<th>Reference</th>
</tr>
</thead>
<tbody>
<tr>
<td>Point Beach assembly</td>
<td>Flooded</td>
<td>0.25808</td>
<td>SNF-4875</td>
</tr>
<tr>
<td>Calvert Cliffs full PWR assembly</td>
<td></td>
<td>0.24423</td>
<td>SNF-4875</td>
</tr>
<tr>
<td>Calvert Cliffs partial PWR assembly</td>
<td></td>
<td>0.25966</td>
<td>SNF-4875</td>
</tr>
<tr>
<td>Point Beach assembly</td>
<td>X</td>
<td>0.947</td>
<td>SNF-4875</td>
</tr>
<tr>
<td>Calvert Cliffs full PWR assembly</td>
<td>X</td>
<td>0.937</td>
<td>SNF-4875</td>
</tr>
<tr>
<td>Calvert Cliffs partial PWR assembly</td>
<td>X</td>
<td>0.948</td>
<td>SNF-4875</td>
</tr>
<tr>
<td>Loose fuel pins</td>
<td></td>
<td>0.161</td>
<td>HNF-4832</td>
</tr>
<tr>
<td>Loose fuel pins</td>
<td>X</td>
<td>0.630</td>
<td>HNF-4832</td>
</tr>
<tr>
<td>Loose fuel pins (plus no inner pipe)</td>
<td>X</td>
<td>0.976</td>
<td>HNF-4832 (double contingent case)</td>
</tr>
</tbody>
</table>


NAC = Nuclear Assurance Corporation.
PWR = pressurized water reactor.
To validate the KENO calculations, critical benchmark experiments were chosen that used UO₂ fuel pins with enrichments bounding those for the LWR fuel pins loaded in the NAC-1 casks. From the comparison of KENO-Va results to the benchmark experiments, the code bias in $k_{\text{eff}}$ was determined to be 0.004, with a standard deviation of 0.007. This bias was included in the upper bound $k_{\text{eff}}$ values in Table D6-6. Also, use of the XSDRN code for determining the most reactive configurations was validated by direct comparison to KENO-Va for optimization of the Point Beach fuel, single cask analysis. Both codes indicated that the 179 pin configuration was the most reactive.

NFS-4/NAC-1 Spent Fuel Shipping Cask Criticality Safety Evaluation Report

HNF-4832, CSER 99-004: NFS-4/NAC-1 Spent Fuel Shipping Cask Criticality Safety Evaluation Report, analyzes the $k_{\text{eff}}$ of a NAC-1 cask loaded with loose fuel pins from the 324 Building B-Cell Safety Cleanout Project. 118 loose fuel pins are to be loaded into a stainless steel pipe intended to restrict the pin spacing for the purpose of criticality safety. To ease the loading of the inner pipe containing the pins, the pipe will be cut lengthwise into two half cylinders. After loading, a steel plate will be welded over each half cylinder and the two half cylinders will be secured together for insertion into the NAC-1 cask. The $k_{\text{eff}}$ analysis assumed that the inner pipe containing the fuel pins has a circular cross-section and the steel plate welded to the half cylinders was neglected giving a more reactive configuration in the model than in the actual design.

The materials of the NAC-1 cask were explicitly modeled, including the inner cylinder of stainless steel, the lead shielding, the outer stainless steel of the neutron shield tanks, and the top and bottom of the casks. The fuel was modeled with the following assumptions:

- Unirradiated fuel was modeled; no credit was taken for burnup
- Loading of the cask is limited to 17 PWR pins with enrichment $< 3.04\%$
- Loading of the cask is limited to 101 BWR pins with enrichment $< 2.93\%$
- The zircaloy cladding was included in the model.

The $k_{\text{eff}}$ analysis considered a normal dry cask configuration with the inner pipe containing close-packed, dry fuel pins. Water reflection around the cask was assumed. The analysis also considered abnormal flooded conditions with water inside the cask and inner pipe. In addition, abnormal configurations were analyzed for a flooded cask without the inner pipe constraining the fuel. In these cases, fuel pitch within the inner cask region was varied to determine the optimal $k_{\text{eff}}$. The abnormal cases with no inner pipe were included to examine whether or not the inner pipe constraining the fuel geometry was required to fulfill double contingency for a flooded cask (flooding being the first contingency).

The MCNP computer code was used to determine $k_{\text{eff}}$. A series of benchmark experiments for low-enriched uranium systems was used to determine that the bias in $k_{\text{eff}}$ for this analysis was 0.004. A standard value of 0.005 was used for the standard deviation in $k_{\text{eff}}$ due to input uncertainties (e.g., dimensional tolerances, enrichment error, etc.). These uncertainties
were combined with the statistical error in the Monte Carlo population to determine an error in the calculated $k_{\text{eff}}$ of 0.011, including bias and uncertainty at the 95% confidence level. This value was added to all calculated values of $k_{\text{eff}}$ and the total was reported in Table D6-6.

**D6.3.3.3.2 Analysis Results.** Analysis results for $k_{\text{eff}}$ of the NAC-1 casks are summarized in Table D6-6. The table contains entries for all cask configurations and all credible hazardous conditions.

**Normal Conditions**

Normal conditions for the NAC-1 casks consist of five casks with dry intact fuel in ISO containers stored side-by-side on the ISA pad. The most reactive configuration for dry cask arrays was found to have $k_{\text{eff}} = 0.25966$ for the Calvert Cliffs partial assembly in an infinite planar array. From the parametric study of optimal moderation, a completely dry array was found to be most reactive. This result represents an upper bound on $k_{\text{eff}}$ for the actual NAC-1 cask array with three Calvert Cliffs assemblies, two Point Beach assemblies, and one cask loaded with odd pins. The normal configuration for the NAC-1 casks has a $k_{\text{eff}}$ well below the safe subcritical limit of 0.95.

**Abnormal Conditions**

The only unlikely but credible hazardous event for the NAC-1 casks is flooding, which is presented below as a single unlikely contingency event.

**Single-Contingency Hazardous Conditions**

**Flooding** – From Table D6-6, the most reactive flooded configuration of NAC-1 casks was found to be an infinite planar array of Calvert Cliffs partial assemblies that resulted in $k_{\text{eff}} = 0.948$, including bias and uncertainty at the 95% confidence level. This array represents an upper bound on the $k_{\text{eff}}$ of the actual NAC-1 cask configuration. Thus, the $k_{\text{eff}}$ of all credible single-contingency hazardous configurations for the NAC-1 casks falls below the safe subcritical limit of 0.95.

**Other Credible Hazardous Conditions**

For the loose fuel pins, a flooded cask with no inner pipe constraining the geometry of the fuel resulted in an upper bound on $k_{\text{eff}}$ of 0.976. This credible hazardous condition exceeds the safe subcritical limit of 0.95. To control this hazard, all fuel must be constrained within a cylindrical geometry with a diameter no greater than 10.9 in. within the cask, and the component(s) or structure providing this constraint must be qualified to maintain its structural integrity to the same degree as the casks. With fuel in the inner pipe (rod container), the maximum $k_{\text{eff}}$ for the system is 0.630, well below the safe subcritical limit of 0.95. This control is discussed further in Section D6.4 and Chapter D4.0.
D6.4 CRITICALITY CONTROLS

D6.4.1 Engineered Controls

D6.4.1.1 Fast Flux Test Facility Fuel. The geometry of the CCC was assumed to be intact in the criticality hazard analyses. Thus, no \( k_{\text{eff}} \) analyses were performed for double batching (the CCC geometry is assumed to prevent this hazard) or rearrangements of the CCC tubes due to structural failure. To protect the assumptions in Section D6.3.3 for the \( k_{\text{eff}} \) analysis of hazardous conditions, the ISC and the CCC are designated as system geometry controls that must be designed to maintain structural integrity for all design basis accidents (e.g., dropping, collision, etc.), including the seismic event. The design classification of the CCC and ISC structures and their detailed design requirements are discussed in Chapter D4.0.

D6.4.1.2 Commercial Light Water Reactor Fuel. From the results of the analysis in HNF-1832, the geometry constraint of the inner pipe containing the loose fuel pins is required in order to provide double-contingency protection against criticality due to flooding of the cask. Therefore, the inner pipe is a safety-class component that must be designed and structurally qualified to the same quality level as the cask. The design classification of the inner pipe containing the fuel and its detailed design requirements are discussed in Chapter D4.0.

The square lattice geometry of the fuel assemblies was assumed to be intact by the analysis in SNF-4875. Thus, no \( k_{\text{eff}} \) analyses were performed for fuel rubble or rearrangements of fuel within the inner canister due to structural failure. To protect the assumptions in Section D6.3.3 for the \( k_{\text{eff}} \) analysis of hazardous conditions, the NAC-1 cask, inner container, and fuel assembly are designated as system geometry controls that must be designed and maintained to ensure structural integrity for all design basis accidents (e.g., dropping, collision, etc.), including the seismic event. The design classification of the cask and fuel assembly structures and their detailed design requirements are discussed in Chapter D4.0.

D6.4.2 Administrative Controls

D6.4.2.1 Fast Flux Test Facility Fuel. The Identi-69 analysis presented in WHC-SD-FV792-DA-004 makes assumptions regarding the fuel form and enrichment that are protected by administrative controls to ensure that all ISC configurations remain within the analyzed envelope. Specifically, ISCs containing fuel pins having \( >29.28 \) wt\% Pu or any other experimental pins not analyzed in Section D6.3.3 cannot be accepted at the ISA without further analysis.

The criticality analysis for metal fuel assemblies in WHC-SD-FF-CSER-004 (Rev. 1-B) did not systematically analyze mixtures of Identi-69 containers with metal assemblies in a CCC. Also, the CSER did not analyze certain misloaded ISCs containing metal assemblies structurally compromised by hot-cell rot. To ensure that configurations not analyzed by the CSER cannot
exist at the ISA, acceptance of ISCs containing metal assemblies is prohibited by an administrative control until further analysis has been performed on metal assemblies with hot-cell rot and with mixtures of Ident-69 containers and metal assemblies.

The criticality analysis in WHC-SD-FF-CSER-004 (Rev. 1-A) for repackaged fuel material and experimental pins assumed that not more than 4 kg of repackaged fuel and pins were contained in the spent fuel container loaded in a modified Ident-69 container. Also, as discussed in Section D6.3.3, the analysis considered only experimental pins that have been classified as MOX fuel, with not more than 31 wt% Pu enrichment. To ensure that the existing analyses envelope all ISC configurations, these assumptions are protected by administrative controls. Specifically, the following ISC configurations cannot be accepted at the ISA without further analysis:

- ISCs containing a modified Ident-69 with more than 4 kg of repackaged fuel material and experimental pins.
- ISCs containing repackaged fuel material and experimental pins that are not packaged in stainless steel tubes packed within a spent fuel container inside a modified Ident-69, as described in Section D6.3.2.
- ISCs with experimental fuel not classified as MOX or having greater than 31 wt% Pu.
- ISCs containing fuel pieces that may be mounted in moderating material (e.g., plastic).

The criticality analysis results in Section D6.3.3 indicate that a failure of administrative controls at FFTF, resulting in any other type (i.e., other than misloading a modified Ident-69 with experimental fuel and repackaged fuel, or misloading of an ISC with metal assemblies as described previously) of misloaded ISC that is transferred to the ISA, will not create a criticality concern at the ISA. All other credible accident scenarios associated with misloading at FFTF were shown to be safely subcritical and satisfy the double-contingency principle, without the requirement for additional administrative controls on cask receipt at the ISA.

**D6.4.2.2 Commercial Light Water Reactor Fuel.** The criticality analysis for loose fuel pins showed that it is necessary to load the fuel into the inner pipe geometry for double-contingency protection against flooding. Therefore, a NAC-1 cask with loose fuel pins cannot be accepted at the 200 Area ISA unless the pins are loaded with an inner pipe configuration, as described in Chapter D2.0.
D6.4.3 Application of Double Contingency

A general discussion regarding how the double-contingency principle is applied is provided in Section 6.4.3 of the SNF Project FSAR.

D6.4.3.1 Fast Flux Test Facility Fuel. Table D6-4 shows the largest $k_{eff}$ found in the analysis for normal cask configurations was 0.45 for five metal fuel assemblies and two DFAs in the CCC. Thus, all normal cask configurations have an upper bound $k_{eff}$ well below the safe subcritical limit of 0.95.

For those hazardous configurations that result from, at most, a single unlikely contingency event (i.e., abnormal conditions that do not satisfy double contingency, such as misloading of a dry cask, etc.), the maximum $k_{eff}$ is found to be 0.9494 for a CCC loaded with seven DFAs that have experienced complete failure due to hot-cell rot. Thus, all abnormal cask configurations that do not result from two or more unlikely, independent events have an upper bound $k_{eff}$ below the safe subcritical limit of 0.95.

Some of the hazardous configurations in Table D6-4 that result from two or more unlikely contingency events have an upper bound on $k_{eff}$ that exceeds the safe subcritical limit of 0.95. Notably, the hazardous configuration with a flooded and tipped CCC loaded with seven DFAs that have failed due to hot-cell rot was found to have a $k_{eff}$ upper bound of 1.155. Not all hazardous configurations resulting from two or more unlikely events were analyzed in the CSERs, so there may be other conceivable hazardous configurations that have an even higher upper bound on $k_{eff}$, as calculated by the analysis scheme in the CSERs. Since these cases already satisfy the double-contingency principle, further controls to prevent criticality are not required in order to satisfy the double-contingency principle.

D6.4.3.2 NRF TRIGA Fuel. Table D6-5 shows the largest $k_{eff}$ found in the analysis for normal cask configurations was 0.5818 for stainless steel fuel assemblies in the Rad-Vault. Thus, all normal cask configurations have an upper bound $k_{eff}$ well below the safe subcritical limit of 0.95.

The maximum $k_{eff}$ for single-contingency hazardous events affecting the TRIGA casks was found to be 0.858 for the crushed Rad-Vault configuration represented by the infinite fuel assembly array analyzed in WHC-SD-FF-CSER-006. Thus, all abnormal Rad-Vault configurations that do not result from two or more unlikely, independent events have an upper bound $k_{eff}$ that is below the safe subcritical limit of 0.95.

One hazardous condition analyzed in WHC-SD-FF-CSER-006 resulting from two or more unlikely contingency events (e.g., crane failure, rubble formation in optimal geometry, and separation of fuel from the cask stainless steel) was found to have an upper bound on $k_{eff}$ exceeding the safe subcritical limit of 0.95. Since this case already satisfies the double-contingency principle, further controls to prevent criticality are not required in order to satisfy the double-contingency principle.
D6.4.3.3 Commercial Light Water Reactor Fuel. Table D6-6 shows that the largest $k_e$ found for normal cask configurations was 0.25966 for the Calvert Cliffs partial assembly. Thus, all normal NAC-I cask configurations have an upper bound $k_e$ well below the safe subcritical limit of 0.95.

The maximum $k_e$ for single-contingency hazardous events affecting the NAC-1 casks was found to be 0.948 for the flooded Calvert Cliffs partial assembly. Thus, all abnormal NAC-1 cask configurations that do not result from two or more unlikely, independent events have an upper bound $k_e$ that is below the safe subcritical limit of 0.95.

One double-contingent event analyzed in HNF-4832 (i.e., flooded cask with loose fuel pins not constrained by an inner pipe) was found to have an upper bound on $k_e$ exceeding the safe subcritical limit of 0.95. The pipe containing the fuel within the inner cask was designated as a geometry control to ensure that double contingency is satisfied for NAC-1 cask flooding events. With the pipe intact, $k_e$ was analyzed to be below the safe subcritical limit of 0.95.

D6.5 CRITICALITY PROTECTION PROGRAM

Section 6.5 of the SNF Project FSAR provides an overview of the organizational structure and interfaces and the technical and administrative practices of the criticality protection policy and programs that are being developed for the SNF Project operations.

D6.6 CRITICALITY INSTRUMENTATION

This section addresses the need for a criticality alarm system and a criticality detection system in the ISA. DOE Order 5480.24 references ANSI/ANS-8.3, Criticality Accident Alarm System, for requirements relating to nuclear criticality alarm systems. ANSI/ANS-8.3 states that neither a criticality alarm system or criticality detection system is required, where the probability of a criticality accident is determined to be less than $1 \times 10^{-6}$ per year. Interpretive guidance on the probability determination (Holten 1993) states: "The use of $10^{-6}$ does not necessarily mean that a PRA [probabilistic risk assessment] has to be performed. Reasonable grounds shall be presented on the basis of commonly accepted engineering judgement." Accordingly, the remaining discussion supports the judgement that no criticality alarm or detection systems are required in the ISA.

The hazard evaluation in Section D6.3.2 has identified credible hazardous conditions related to criticality at the ISA. The results of the criticality analyses (Section D6.3.3) show that with the engineered and administrative controls identified in Section D6.4.2, all credible hazardous conditions at the ISA are safely subcritical. Therefore, no criticality alarm system or criticality detection system is required at the ISA. The engineered and administrative controls in Section D6.4.2 are required to: (1) ensure that no unanalyzed hazardous configurations can exist
at the ISA, and (2) protect the NAC-1 casks from hazardous conditions resulting from flooding of the casks.

**D6.7 REFERENCES**


CHAPTER D7.0

RADIATION PROTECTION
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D7.0 RADIATION PROTECTION

D7.1 INTRODUCTION

The essential features of the radiation protection programs that provide for radiation exposure control, radiological monitoring, and radiological protection instrumentation at all Spent Nuclear Fuel (SNF) Project facilities are addressed in Chapter 7.0 of the SNF Project Final Safety Analysis Report (FSAR). Additional features of radiation protection specific to the 200 Area Interim Storage Area (ISA) are addressed in this Chapter D7.0. The primary function of the 200 Area ISA is to store non-defense reactor spent fuel housed in dry cask storage systems. The purpose of the 200 Area ISA is to consolidate the storage of three types of SNF at a new storage facility in the 200 Area. The fuel types are (1) Fast Flux Test Facility (FFTF) SNF, (2) Neutron Radiography Facility (NRF) Training, Research and Isotope Production, General Atomics (TRIGA) SNF, and (3) 300 Area light water reactor (LWR) SNF. The main radiation protection features of the dry fuel storage systems include: (1) radiation shielding, (2) radioactive material confinement, and (3) minimizing external surface contamination.

A portion of the FFTF SNF is currently stored at the 400 Area ISA within a dry Interim Storage Cask (ISC). The NRF TRIGA SNF is also stored at the 400 Area ISA within a dry storage system consisting of either an NRF TRIGA cask or a U.S. Department of Transportation (DOT)-6M container that is housed within a right-circular cylinder concrete vault (Rad-Vault). These storage systems have been analyzed, reviewed, and approved by the U.S. Department of Energy for storage of these types of fuel at FFTF. The results, although performed for the 400 Area ISA, can be used to assess functional requirements for the storage systems at the 200 Area ISA. The LWR SNF dry storage systems consist of a Nuclear Assurance Corporation (NAC)-1 cask within an International Standards Organization (ISO) container. The LWR storage system is not presently used for SNF storage at FFTF but the system was previously analyzed with the intent to store the LWR fuel at the 400 Area ISA. The results of these analyzes provide information relative to the performance of the storage system that can be used to assess the functional requirements for the LWR SNF storage system at the 200 Area ISA.

The radiological protection features of these dry storage systems are sufficient to preclude on-site or off-site radiological doses in excess of the established requirements (10 CFR 72 and 10 CFR 835) and ALARA (as low as reasonably achievable) goals. This is true for normal storage, off-normal conditions, and accident conditions. No exposures to airborne radioactive material are expected.
D7.2 REQUIREMENTS

The requirements that form the basis for the radiation protection program are identified in Section 7.2 of the SNF Project FSAR.

D7.3 RADIATION PROTECTION PROGRAM AND ORGANIZATION

The SNF Project radiation protection program and its organization, including safety management policies and philosophies, are described in Section 7.3 of the SNF Project FSAR.

D7.4 ALARA POLICY AND PROGRAM

A discussion of the SNF Project ALARA policy and program is provided in Section 7.4 of the SNF Project FSAR. The SNF Project policy regarding ALARA principles during 200 Area ISA design and construction is described in HNF-SD-SNF-PMP-018, Site-Wide Spent Nuclear Fuel Project Management Plan.

Detailed descriptions of the FFTF SNF dry storage system (ISC) shielding configuration are contained in Chapter D4.0. Based on the radiological protection features of the ISC, design analysis results (General Atomics 1995) show that occupational exposure will be well below the ALARA requirements of 10 CFR 72, 10 CFR 835, and HSRCM-1, Hanford Site Radiological Control Manual. The maximum exposure rates allowed by the design are 200 mrem/h at contact on the bottom surface and <2.0 mrem/h for all other surface areas. This higher level for the bottom surface is acceptable because it is normally not accessible. ISC dose rate-shielding analysis calculates the highest accessible surface dose rate to be 1.99 mrem/h at the top of the cask and 99.3 mrem/h at the bottom closure.

A detailed description of the TRIGA/Rad-Vault dry storage system shielding configuration is provided in Chapter D4.0. Based on the radiological protection features of the TRIGA/Rad-Vault, design analysis results (WHC-SD-FF-TI-043) show the highest contact dose rate to be 0.76 mrem/h at the side of the concrete vault. The maximum exposure rate allowed by the design is <5.0 mrem/h at contact with the vault surface.

Detailed descriptions of the LWR SNF dry storage system shielding configuration are contained in Chapter D4.0. Based on the radiological protection features of the NAC-1 cask, design analysis results (HNF-3016) show that occupational exposure will be well below the ALARA requirements of 10 CFR 72, 10 CFR 835, and HSRCM-1. The NAC-1 dose rate-shielding analyses calculate the highest accessible surface dose to be <114 mrem/h. The maximum exposure rates allowed by the design are 200 mrem/h at contact on any surface or 10 mrem/h at 2 meters away from the external surface of the cask.
Routine and accidental releases to the public (off-site receptor) are calculated at the Site boundary shown in Figure D1-2. This is also the location of the controlled area boundary as the term is defined in 10 CFR 72.106. The shielding for all of the dry storage systems also meets the HSRCM-1 requirement for uncontrolled access by the public of <.05 mrem/h (at the ISA fence, see Figure D7-1).

Specific 200 Area ISA design features credited with reducing radiation exposure and achieving ALARA program objectives include the storage system shielding.

**D7.5 RADIOLOGICAL PROTECTION TRAINING**

SNF Project requirements and criteria for radiological protection training are described in Section 7.5 of the SNF Project FSAR.

**D7.6 RADIATION EXPOSURE CONTROL**

A description of SNF Project radiation exposure control measures is provided in Section 7.6 of the SNF Project FSAR. The SNF Project has elected to incorporate in the facility design the following U.S. Nuclear Regulatory Commission (NRC) requirements:

- Apply the radiological exposure criteria of 10 CFR 72.104, “Criteria for Radioactive Materials in Effluents and Direct Radiation from an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage,” to the design and safety analyses.

- Apply the hourly dose limit of 10 CFR 20.1301, “Dose Limits for Individual Members of the Public,” to the design and safety analyses.

- Incorporate control devices in the facility design for access to high-radiation areas that conform to the requirements of 10 CFR 20.1601, “Control of Access to High Radiation Areas.”

For the 200 Area ISA, the radiological exposure annual dose criteria of 10 CFR 72.104 have been incorporated into the storage system designs and safety analysis. These criteria apply to design measures to protect any off-site public individual during normal operations and anticipated occurrences. These annual dose equivalent criteria are 25 mrem (0.25 mSv) to the whole body, 75 mrem (0.75 mSv) to the thyroid, and 25 mrem (0.25 mSv) to any other critical organ.

For the 200 Area ISA, the criteria for the hourly dose limit to the public, as described in 10 CFR 20.1301, have been incorporated into the design and safety analysis. This dose limit (0.002 rem) is assumed to be direct radiation from external sources (casks) for any unrestricted area during normal operations and anticipated occurrences.
During operation, access to the radiological area of the ISA (see Figure D7-1) is controlled using one or more of the methods listed and discussed in Section 7.6.2.6 of the SNF Project FSAR. The potential for the highest dose rates at the ISA occurs in the NAC-1 ISO containers. To avoid unnecessary and inadvertent doses to personnel, the doors to the ISO containers are provided with locks and security seals that can be used to control access. Primarily, access control will be by the use of signs and barricades and the locks on the ISA entry gates and the ISO container doors. Other measures, described in Section 7.6.2.6 of the SNF Project FSAR, will be used if needed.

D7.7 RADIOLOGICAL MONITORING

The radioactive material sampling and monitoring programs conducted within SNF Project facilities are addressed in Section 7.7 of the SNF Project FSAR.

D7.8 RADIOLOGICAL PROTECTION INSTRUMENTATION

A summary of the SNF Project plans and procedures governing radiation protection instrumentation is provided in Section 7.8 of the SNF Project FSAR. Only portable radiation protection instrumentation is used at the ISA.

D7.9 RADIOLOGICAL PROTECTION RECORD KEEPING

Radiological protection record-keeping requirements are described in Section 7.9 of the SNF Project FSAR.

D7.10 OCCUPATIONAL RADIATION EXPOSURES

For the 200 Area ISA, estimated radiation doses to on-site workers were calculated by assessing the operational procedures and planned activities that would result in occupational exposures. Using the estimated amount of time a worker would be near the storage system during handling, surveillance, and maintenance, the estimated occupational exposure to facility workers will be below the 500 mrem/yr administrative control level of HSRCM-1.

Annual surveillance of the ISCs will include visually inspecting the casks and performing a radiological survey in the immediate area of the ISC. Operational analysis indicates that these activities will involve one operator and two radiological control technicians (RCTs), and will require about one hour to complete. Using worst-case dose rates, occupational exposure calculations indicate an annual accumulated dose of approximately 320 mrem to all workers during the inspection of fifty-three casks.
Annual surveillance of the TRIGA casks and the DOT-6M containers will include removing the Rad-Vault lid, visually inspecting the casks and containers, performing a radiological survey, and replacing the lid. Operational analysis indicates that these activities will involve one operator, two RCTs, and support personnel, and will require about two hours to complete. Radiological dose rate calculations for the NRF TRIGA casks were performed using the ISOSHLD Code, conservatively based on 19 maximally exposed fuel elements with an assumed prior irradiation, and an assumed 6-year decay time. (The decay time assumed was from the last powered operation of the TRIGA reactor in May 1989.) Actual cask readings that are representative of the maximum dose rate expected have shown 35 mrem/h (versus 81 mrem/h calculated) on the top and 50 mrem/h (versus 92 mrem/h calculated) on the side. It would naturally follow that the dose at the container could also be expected to be lower than previously calculated. Using worst-case dose rates, occupational exposure calculations indicate an annual accumulated dose of approximately 210 mrem to all workers during inspection of the vault. The exposure to crane operators removing and replacing the lid of the vault would be negligible.

Annual surveillance of the NAC-1 cask will include opening the ISO container door, visually inspecting the cask, and performing a radiological survey. Operational analysis indicates that these activities will involve two operators and one RCT, and will require about two hours to complete. Using worst-case dose rates, occupational exposure calculations indicate an annual accumulated dose of approximately 600 mrem to all workers during inspection of the six casks.

The maximum anticipated exposure resulting from these surveillances and other activities involving lesser exposures is 1150 mrem/year. The majority of this exposure is received by the RCTs. Using an RCT staffing level of four individuals involved in these annual inspections, the average RCT dose per year is 288 mrem total effective dose equivalent, which is well below the 500 mrem (5 mSv) ALARA design goal. The exposure level of all other categories of workers is considerably below this RCT exposure level.

Handling of the ISCs is anticipated to include lifting fixture connection and removal, ISC movement, and installation of the environmental cover. Operational analysis indicates that these activities will involve one rigger and one RCT, and will require about one hour per ISC. These activities primarily require personnel to be near the top of the cask where the exposure is the highest. Using worst case dose rates, occupational exposure calculations indicate an accumulative dose of approximately 210 mrem to all workers during the handling of fifty-three casks.

Based on FFTF Rad-Vault loading experience, the estimated total personnel exposure time to load the Rad-Vault with two DOT-6M containers and six TRIGA casks is approximately 4 hrs. Operational analysis indicates that these activities will involve one operator and one RCT, and that the exposure levels experienced are similar to those involved with the Rad-Vault surveillance described above (i.e., each occurs with the Rad-Vault lid removed and with personnel above the open storage casks). Occupational exposure calculations indicate an accumulative dose of approximately 280 mrem to all workers during vault loading.
Handling of the NAC-1 casks is anticipated to primarily include inspection of the cask and ISO container after it is in place, since the connection and disconnection of the handling fixture is remotely operated. Operational analysis indicates that this activity will involve one operator and one RCT, and will take less than 15 minutes to complete. Using worst-case dose rates, occupational exposure calculations indicate an accumulative dose of approximately 300 mrem to all workers during handling of six casks.

The maximum anticipated exposure resulting from these handling and loading activities is 790 mrem/yr. Using a conservative staffing level of four individuals, the average worker dose per year is 197 mrem total effective dose equivalent, which is well below the 500 mrem (5 mSv) ALARA design goal. It is not anticipated that cask handling and vault loading will occur in the same year as system surveillances; therefore, only one of the anticipated annual exposures needs to be considered in evaluating compliance to the ALARA goal.

D7.11 REFERENCES


Figure D7-1. Radiological Control Boundary.
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CHAPTER D8.0

HAZARDOUS MATERIAL PROTECTION
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D8.0 HAZARDOUS MATERIAL PROTECTION

D8.1 INTRODUCTION

The major provisions of the occupational safety and health program, as the program applies to hazardous material protection for the Spent Nuclear Fuel (SNF) Project, are addressed in Chapter 8.0 of the SNF Project Final Safety Analysis Report (FSAR).

D8.2 REQUIREMENTS

The requirements that form the basis for the hazardous material protection program are identified in Section 8.2 of the SNF Project FSAR.

D8.3 HAZARDOUS MATERIAL PROTECTION PROGRAM AND ORGANIZATION

The SNF Project has established a visible and comprehensive occupational safety and health program. This program is described in Section 8.3 of the SNF Project FSAR.

D8.4 ALARA POLICY AND PROGRAMS

While there is no established formal SNF Project as-low-as-reasonably-achievable (ALARA) program for nonradiological hazardous materials, the SNF Project has expanded the classic concept of ALARA (i.e., minimization of radiological exposures) to the application of exposure minimization for hazardous substances and conditions. The SNF Project’s policy is described in Section 8.4 of the SNF Project FSAR.

Work practices for hazardous material protection and control of chemical exposures, as stated in Section 8.4 of the SNF Project FSAR, will be implemented at the 200 Area Interim Storage Area (ISA) using approved SNF Project implementing procedures. The occupational safety and health program will use the additional provisions of Section 8.4 of the SNF Project FSAR. These provisions will also be implemented at the ISA using approved SNF Project implementing procedures.

Applicable ergonomics considerations based on DOE Order 5480.10, Contractor Industrial Hygiene Program (Section 9B), under the occupational safety and health program, industrial hygiene subprogram, are included in Chapter D13.0. Ergonomics considerations, along with risk factor control processes and the ergonomics program, are described in the SNF Project human engineering program plan (see Section 8.4 of the SNF Project FSAR and
SNF-4399). This document will be revised as necessary to address integration of this plan with HNF-MP-003, Integrated Environment, Safety and Health Management System Plan.

**D8.5 HAZARDOUS MATERIAL TRAINING**

Plans and procedures for training SNF Project workers regarding hazardous materials are summarized in Section 8.5 of the SNF Project FSAR.

Section 8.5 of the SNF Project FSAR states that SNF Project management provides training, professional education, and certification opportunities necessary to support, maintain, and enhance industrial hygiene staff proficiency to meet or exceed U.S. Department of Energy industrial hygiene training objectives and goals in accordance with the SNF Project Standards/Requirements Identification Document (HNF-SD-SNF-RD-001). ISA management is responsible for ensuring that workers assigned to any task involving hazardous materials are trained in the safety and health hazards associated with such hazardous materials. Workers will perform only those tasks for which they have received the proper training. If the ISA mission changes, ISA management will review the training requirements and modify them accordingly. ISA management is also responsible for ensuring that retraining is provided within the time allowed by training course requirements.

**D8.6 HAZARDOUS MATERIAL EXPOSURE CONTROL**

Worker safety features at the ISA are an integral part of facility design and operation. The ISA design encompasses human factors considerations to ensure that operations can be conducted safely. SNF Project occupational exposures to hazardous materials and the spread of hazardous material contamination are controlled by a combination of engineered, operational, and administrative controls, and by the use of personal protective clothing and equipment. These controls are described in Section 8.6 of the SNF Project FSAR.

Construction of the ISA has been performed to minimize the use of hazardous materials and the generation of hazardous and non-hazardous waste. No polychlorinated biphenyls or asbestos will be used. Transformers, lighting, and other electrical equipment that use an insulating oil will be polychlorinated biphenyl-free.

No significant hazardous materials have been identified for the ISA as a result of a hazard analysis that was performed and documented in SNF-4820, 200 Interim Storage Area Final Hazard Analysis Report, except for the radionuclide content in the dry cask storage systems. HNF-2524, 200 East Area Interim Storage Area Preliminary Safety Evaluation Report, states that these materials are primarily uranium oxide and mixed uranium and plutonium oxide, which are known to have toxicological effects. As stated in Section 3.1.1 of HNF-2524, the toxicological hazards of the radionuclide inventory were found to be bounded by the radiological consequences.
Small quantities of other hazardous material identified by the hazard identification process include pyrophoric metals and hydrides, oxidizers, hydrogen, diesel fuel, and other flammable or combustible materials. No routine chemical processes will be conducted in the ISA. Some chemicals, such as those used for equipment decontamination, may be used occasionally. Table D3-1 identifies hazards by form, type, location, and total quantity.

The FFTF SNF inventory includes a small amount of metallic sodium inside the sodium-bonded fuel pins. Purging and backfilling of pressure monitoring equipment will involve the use of an inert gas. In addition, some chemicals, such as those used for equipment decontamination, may be used occasionally.

An inert atmosphere (helium and argon) is maintained in storage containers for FFTF SNF. This inherently precludes a potential hazard of pyrophoric uranium or plutonium reacting with air (oxygen) that may have leaked in the SNF container. In addition, there is a potential for the sodium to react with air (oxygen). Potential for this event is low since only 8 of 329 FFTF items are metallic fuel; the remainder are not pyrophoric. In addition, the inert atmosphere will preclude reaction of air with either the metallic uranium or sodium. These eight items will be consolidated into two casks.

Major features of worker protection are presented in Table D3-4 and are categorized by hazard. These features are in addition to safety-class or safety-significant features for design basis accidents. No safety-significant structures, systems, and components or Technical Safety Requirements have been identified for the ISA based solely on worker safety considerations.

All work activities in the ISA will receive adequate advance planning so that if potential hazardous materials are identified due to changing conditions in the future, specific precautions will be applied. The exposure controls identified in Sections 8.6.1 through 8.6.4 of the SNF Project FSAR will then be implemented at the ISA using approved SNF Project implementing procedures.

Other hazardous materials at the ISA (maintenance materials) will be properly inventoried and stored to control hazards inherent to the material, in accordance with SNF Project implementing procedures.

**D8.7 HAZARDOUS MATERIAL MONITORING**

Summaries of the hazardous material sampling and monitoring programs that are conducted internally and externally for SNF Project facilities are provided in Section 8.7 of the SNF Project FSAR. The workplace and external monitoring program described in Sections 8.7.1 and 8.7.2 of the SNF Project FSAR will be implemented at the ISA, as appropriate, using approved SNF Project implementing procedures. An environmental, radioactivity, and chemical emissions monitoring program, including requirements, is presented in Section 8.7.2 of the

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Surveillance and monitoring activities to ensure safe storage of SNF at the ISA are presented in Section D2.3.2. These activities include the following:

- Annual surveillance of the interim storage casks (ISC) – This surveillance involves a visual inspection, radiation survey and smear sampling about the ISC environmental cover.

- Annual surveillance of Training, Research and Isotope Production, General Atomics (TRIGA) fuels in U.S. Department of Transportation (DOT)-6M containers – This surveillance includes a visual inspection and radiation surveys of the Rad-Vault, visual inspection and radiation survey of the fuel cask, and smear sampling of the fuel casks.

- Annual surveillance of the Nuclear Assurance Corporation (NAC)-1 casks in International Standards Organization (ISO) containers – The surveillance includes a visual inspection and radiation survey of the ISO container external surface, radiation survey and smear samples of the top of NAC-1 casks, and visual inspection of the interior of the ISO container and exterior of the NAC-1 casks.

D8.8 HAZARDOUS MATERIAL PROTECTION INSTRUMENTATION

Summaries of plans and procedures governing hazardous protection instrumentation are provided in Section 8.8 and Table 8-2 of the SNF Project FSAR. As stated in Section 8.8 of the SNF Project FSAR, safety and health specialists will determine the need for hazardous protection instrumentation and the number and placement of instruments under normal and emergency conditions, in accordance with the requirements stated in Section 8.8 of the SNF Project FSAR.

D8.9 HAZARDOUS MATERIAL PROTECTION RECORD KEEPING

The SNF Project has an established document control and records management program. This program is summarized in Section 8.9 of the SNF Project FSAR.

D8.10 HAZARD COMMUNICATION PROGRAM

The hazard communication program applies to the purchase, receipt, transportation, use, and storage of hazardous chemicals and products. This program is summarized in Section 8.10 of the SNF Project FSAR. The hazard communication program for the ISA will be implemented in accordance with the provisions of Sections 8.10.1 through 8.10.6 of the SNF Project FSAR, including hazard posting in work areas, chemical management, chemical labeling, chemical
product list, material safety data sheets, information and training. This program will be accomplished in accordance with approved SNF Project implementing procedures.

**D8.11 OCCUPATIONAL CHEMICAL EXPOSURES**

Predicted annual exposures to workers from hazardous material sources are identified in Section 8.11 of the SNF Project FSAR. The identification of chemical hazard locations, posting, chemical management, and other controls to limit occupational chemical exposure is included in Section 8.10 of the SNF Project FSAR.

**D8.12 REFERENCES**


CHAPTER D9.0

RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT
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D9.0 RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT

D9.1 INTRODUCTION

The essential features of the radioactive and hazardous waste management programs that provide for the safe control, collection, and handling of wastes generated during routine operations at Spent Nuclear Fuel (SNF) Project facilities are detailed in Chapter 9.0 of the SNF Project Final Safety Analysis Report (FSAR). This Chapter D9.0 only applies to waste generated within the 200 Area Interim Storage Area (ISA) and those systems designed to deal with that waste. The 200 Area ISA will store SNF in dry cask storage systems. No uncontained radioactive materials will be handled. Spent fuel in shipping containers is received, unloaded, and stored at the 200 Area ISA. No chemical, toxicological, or hazardous materials will normally exist on-site except for the SNF within the storage systems. Confinement of the 200 Area ISA radioactive materials is a design feature of each dry spent fuel storage system.

D9.2 REQUIREMENTS

The requirements that form the basis for the radioactive and hazardous waste management program are found in Section 9.2 of the SNF Project FSAR.

D9.3 RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT PROGRAM AND ORGANIZATION

The facility administrative procedures for solid waste management contain the procedural guidance for the planning, generation, and disposal of generated waste in compliance with applicable requirements. The administrative procedures cover characterization, preplanning, designation, containerization, disposal, and programmatic requirements. A summary of the SNF Project waste management program is provided in Section 9.3 of the SNF Project FSAR.

D9.4 RADIOACTIVE AND HAZARDOUS WASTE STREAMS AND SOURCES

The only waste streams at the ISA are solid wastes; primarily light bulbs, vegetation and animal carcasses, and waste generated during contamination monitoring. The estimated volumes for these wastes are:

- Sodium light bulbs - < 1 L/yr (0.25 gal/yr) (recycled)
Miscellaneous solid waste (primarily rags, wipes, protective clothing, vegetation, and carcasses) - < 1 m³/yr (35 ft³/yr) (potentially radioactive).

In the unlikely event of a radioactive release, cleanup materials will be designated as low-level waste. Since the 200 Area ISA is not manned, no sanitary sewage is generated. The low-level waste will be packaged per the Waste Management Federal Services of Hanford, Incorporated waste acceptance requirements (WHC-EP-0063-5)—described in Section 16 of the SNF Project Standards/Requirements Identification Document (HNF-SD-SNF-RD-001) and in facility waste administrative procedures—and then transported to Waste Management Federal Services of Hanford for disposition. Sodium light bulbs will be delivered to the 400 Area consolidation area where they will be picked up by commercial treatment and disposal facility operators. Miscellaneous nonradioactive waste will be disposed of at an off-site waste disposal site.

D9.5 REFERENCES


CHAPTER D10.0

INITIAL TESTING, IN-SERVICE SURVEILLANCE, AND MAINTENANCE
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D10.0 INITIAL TESTING, IN-SERVICE SURVEILLANCE,
AND MAINTENANCE

D10.1 INTRODUCTION

Essential features of the initial testing program, operational readiness review, in-service surveillance program, and the maintenance program implemented at Spent Nuclear Fuel (SNF) Project facilities are described in Chapter 10.0 of the SNF Project Final Safety Analysis Report (FSAR). 200 Area Interim Storage Area (ISA)-specific features of these programs are described in this chapter.

D10.2 REQUIREMENTS

The requirements that form the basis for the initial testing, surveillance, and maintenance programs are found in Section 10.2 of the SNF Project FSAR. Specific requirements applicable to this chapter include the following:

- U.S. Department of Energy (DOE) O 430.1A, Life Cycle Asset Management
- DOE Order 4330.4B, Maintenance Management Program
- DOE Order 5480.19, Conduct of Operations Requirements for DOE Facilities.

D10.3 INITIAL TESTING

The SNF Project initial testing program ensures the operability of equipment and facilities before facility operation. Project details of this program are provided in Section 10.3 of the SNF Project FSAR. Due to the simple passive nature of the facility and its components, the 200 Area ISA initial testing program will consist entirely of construction acceptance testing. No special testing requirements prior to receipt of SNF have been identified for the 200 Area ISA systems and components.

D10.4 IN-SERVICE SURVEILLANCE PROGRAM

The SNF Project in-service surveillance program is designed to maintain the integrity of facility systems and to ensure that systems perform their function of protecting the health and safety of the public, workers, and facility staff by preventing or mitigating accident consequences. Details of this program are provided in Section 10.4 of the SNF Project FSAR. The Technical Safety Requirements described in Chapter D5.0 identify no safety surveillance requirements for the 200 Area ISA.
Periodic inspection of the interim storage casks (ISCs) is required to support analysis assumptions. This annual surveillance will include a visual inspection, radiation survey, and smear sampling about the ISC environmental cover.

Annual surveillance of the Training, Research and Isotope Production, General Atomics (TRIGA) cask and U.S. Department of Transportation (DOT)-6M containers will include (1) visual inspection and radiation survey of the Rad-Vault, (2) vault lid removal to allow visual inspection and radiation surveys of the fuel casks/containers, and (3) smear sampling of the fuel casks/containers. Since the Rad-Vault is constructed of concrete, an annual visual inspection of the underside is not required.

The seal integrity program for TRIGA fuel storage and related preventive maintenance and/or testing requirements shall be identified for TRIGA fuel containers prior to year 2015 to protect seal design life considerations.

Annual surveillance of the Nuclear Assurance Corporation (NAC)-1 casks in International Standards Organization (ISO) containers will include (1) a visual inspection of the external surface of the ISO container, (2) radiation surveys of the exterior surface of the ISO, (3) a radiation survey of the top of the NAC-1 casks upon opening the ISO container doors, (4) smear samples from the top of the NAC-1 casks, and (5) visual inspections of the interior of the ISO container and exteriors of the NAC-1 casks. The ISO has structural members constructed of ferrous materials, and an inspection of the underside is appropriate every 5 years (to maintain ISO certification).

D10.5 MAINTENANCE PROGRAM

The maintenance program for the SNF Project facilities is conducted in accordance with DOE Order 4330.4B, Maintenance Management Program, which provides the general policy and objectives for establishing cost-effective maintenance and repair programs for DOE property. The maintenance program will incorporate the results of considerable project and subsystem vendor interface activities aimed at ensuring an acceptable design and acceptable operating practices relative to reliability, availability, and maintainability.

Policies and procedures are in place to effectively manage SNF Project facility maintenance activities. Section 10.5 of the SNF Project FSAR summarizes the maintenance policies and procedures that are implemented at the 200 Area ISA. The 200 Area ISA facility components are classified as General Service. The dry cask systems have been designated Safety Significant. Due to the simple passive nature of the facility and its components, minimal maintenance activities have been identified by the SNF Project for this facility. Identified maintenance tasks include monitoring and maintaining the equipment for receipt of SNF storage systems. During long-term storage, maintenance tasks include painting of the storage cask systems, lamping, fence and gate inspections and repairs, and vegetation removal. The Viton
O-rings on the Neutron Radiography Facility (NRF) TRIGA cask have a 20-year design life (per vendor) and require evaluation prior to year 2015.

**D10.6 REFERENCES**


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CHAPTER D11.0

OPERATIONAL SAFETY
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**D11.0 OPERATIONAL SAFETY**

**D11.1 INTRODUCTION**

Features of the Spent Nuclear Fuel (SNF) Project conduct of operations and fire protection programs are described in the following sections and in Chapter 11.0 of the SNF Project Final Safety Analysis Report (FSAR).

**D11.2 REQUIREMENTS**

The requirements that establish the basis for conduct of operations and general aspects of operational safety are identified in Section 11.2 of the SNF Project FSAR. Specific requirements applicable to this chapter include the following:

- DOE Order 5480.7A, *Fire Protection*
- RLID 5480.7, *Fire Protection*

**D11.3 CONDUCT OF OPERATIONS**

"Conduct of operations" is a set of principles that establishes an overall philosophy for achieving excellence in the operation of the SNF Project facilities. SNF Project application of conduct of operations principles is described in Section 11.3 of the SNF Project FSAR.

**D11.4 FIRE PROTECTION**

The fundamental fire protection programs for SNF Project facilities are addressed in Section 11.4 of the SNF Project FSAR. The elements of the fire protection program that are specific to the 200 Area Interim Storage Area (ISA) are described in the following subsections. The results of a 200 Area ISA fire hazard analysis (FHA), documented in SNF-4932, *Fire Hazard Analysis for the 200 Area Interim Storage Area*, are addressed in Section D3.3. In addition, a FHA was performed for the proposed ISA storage building and the results are documented in HNF-4552, *Preliminary Fire Hazards Analysis for the 200 East Area Interim Storage Area Storage Building*. Results from these analyses are summarized in the following subsections.
D11.4.1 Fire Hazards

The FHAs (SNF-4932, HNF-4552) comprehensively assessed the risk from fire at the 200 Area ISA and storage building to determine that: (1) the potential for occurrence of fire is minimized, (2) a fire would not cause an on-site or off-site release of radiological and other hazardous materials that would threaten the public health and safety or the environment, (3) requirements are in place that will provide an acceptable degree of life safety to SNF Project and contractor workers, and (4) the safety systems are not damaged by fire.

SNF-4932 and HNF-4552 were prepared to meet the requirements of U.S. Department of Energy (DOE) Order 5480.7A and to evaluate compliance to DOE fire protection criteria. As required, this analysis addressed the following elements:

- Description of construction
- Protection of essential safety-class equipment
- Fire protection features
- Description of fire hazards
- Life safety considerations
- Critical process equipment
- High value property
- Damage potential; maximum credible fire loss and maximum possible fire loss
- Fire department or brigade response
- Recovery potential
- Potential for toxic, biological, and/or radiological incident due to a fire
- Emergency planning
- Security considerations related to fire protection
- Natural hazards (earthquake, flood, wind) impact on fire safety
- Exposure fire potential, including the potential for fire spread between fire areas.

Future changes to the FHAs will be screened to these elements to ensure that no unreviewed safety question is created.

The ISA facility FHA (SNF-4932) identifies no fire hazards inherent to the 200 Area ISA components, which include interim storage casks (ISCs), Neutron Radiography Facility (NRF) Training, Research and Isotope Production, General Atomics (TRIGA) casks, U.S. Department of Transportation (DOT)-6M containers within a Chem-Nuclear Services, Incorporated Rad-Vault storage vault, and Nuclear Assurance Corporation (NAC)-I casks within International Standards Organization (ISO) containers. Potential fire hazards identified in the FHA include (1) vehicles and equipment required to move the cask storage systems (fuel truck precluded), and (2) combustible materials required during the cask storage system movement activities. Since no credible fire results in damage to the cask storage systems, the maximum possible fire loss (MPFL) is identified as the cost to inspect and evaluate the cask storage systems following exposure to fire. The MPFL value is $100,000. No release of radionuclides to the environment.
from the ISA facility is identified in the FHA, and no toxicological or biological consequences resulting from fire are anticipated. Additional information is provided in the FHA (SNF-4932).

The proposed ISA storage building will be used to store the excessed Fast Flux Test Facility (FFTF) solid waste transfer cask and ISA equipment (e.g., lifting devices, impact limiters, and the tractor-trailer unit). The ISA storage building FHA (SNF-4552) identifies two hazards associated with this building: (1) a fire starting in the tractor-trailer and spreading throughout the facility, and (2) a fire starting in stored combustibles and spreading throughout the facility. The MPFL is identified as the complete loss of the building, with the exception of the FFTF cask. The MPFL value is $3.4 million, including property and cleanup losses. The FFTF cask, which is a one-of-a-kind, high-value property, would have limited damage and could be reused following inspection and evaluation. No release of radionuclides to the environment from the ISA storage building is identified in the FHA as a result of the MPFL event because the cask internals are the only planned component to contain radioactive material. No toxicological or biological consequences resulting from fire are anticipated. Additional information is provided in the FHA (SNF-4552).

D11.4.2 Fire Protection Program and Organization

The fire protection program for SNF Project facilities is structured and implemented in accordance with the operating contractor's safety management policies, philosophies, and criteria described in Section 11.4.2 of the SNF Project FSAR. ISA-specific aspects of the fire protection program are described in the following paragraphs.

The ISC's, NAC-I casks, Rad-Vault, and NRF TRIGA casks have been analyzed to meet the radioactive material release criteria of Title 10, Code of Federal Regulations (10 CFR) Section 71.51(a)(2) from exposure to the 10 CFR Section 71.73(c)(3) fire condition. The FHA for the ISA determined that accidents associated with the ISA are bounded by the transportation fire scenario, which would bound the tractor-trailer fire, mobile crane fire, and runaway fuel truck fire. Therefore, no ISA facility fire protection system is required. Additional information is provided in the FHA (SNF-4932).

One fire hydrant is located 150 ft southwest from the NAC-1/ISO pad (just outside the fence line), and a fire main connection is included in the ISA storage building (just outside the northwest fence line intersection). A pull box is provided at the ISA storage building to facilitate reporting of fires.

Automatic sprinkler protection is provided throughout the ISA storage building and is supplied by a connection to the 200 Area raw water system. The fire protection system is classified as General Service. A fire alarm system is provided that provides for transmission of signals to the Hanford Fire Department and to local building fire alarm annunciators. Portable fire extinguishers will also be available in the ISA storage building.
Although the proposed storage building will not be occupied on a full time basis, adequate life safety features are provided (emergency exit doors and illumination) as required by National Fire Protection Association (NFPA) 101, Life Safety Code. A perimeter gate at the ISA will remain unlocked while personnel are working within the ISA. Based on the open egress point and the absence of significant combustibles, the FHA for the ISA identified no life safety concerns.

**D11.4.3 Combustible Loading Control**

Ordinary combustibles are expected at the ISA facility. The area will be kept free of debris and vegetation, and the perimeter fence will keep transient debris away from the cask storage systems. Combustibles anticipated in the FHA include plywood cribbing, minimal quantities of combustibles to support maintenance activities (fuel truck precluded), and a minimal amount of local vegetation. SNF-4932 found that a combination of plywood and fuel oil was binding to the scenarios evaluated and describes limits on the allowable quantity of plywood cribbing. Ordinary combustibles are expected in the ISA storage building; however, it will not be a general storage structure. Section 11.4.3 of the SNF Project FSAR summarizes the SNF Project program used to prevent unnecessary combustible loadings in project facilities. The transient combustibles control program will be implemented through operating procedures that address the ISA limits established in Technical Safety Requirement AC5.12 (see Section D5.5.3.6).

**D11.4.4 Fire Fighting Capabilities**

The Hanford Fire Department maintains a training program for fire fighting, fire system testing and maintenance, and facility inspections. Fire-fighting capabilities that apply to all SNF Project facilities are addressed in Section 11.4.4 of the SNF Project FSAR. 200 Area ISA-specific fire response procedures are addressed in the following paragraphs.

Due to the absence of any fire detection and alarm system in the 200 Area ISA outside the storage building, personnel action is required to notify the Hanford Fire Department in the event of a fire. No brigade is planned for the ISA and storage building facilities. The standard response to an alarm condition in the 200 Area is by the Hanford Fire Department from the 200 Area fire station. Hanford Fire Department response time from the 200 Area fire station is approximately five minutes. This is the response time and the responder location assumed in the FHAs. WHC-SP-1180, Hanford Site Emergency Response Needs, and HNF-SP-1180, Hanford Site Emergency Response Needs Implementation Plan, provide additional descriptions of the Hanford Fire Department response capabilities. Vehicle access is provided by a paved access road and compacted gravel. The Hanford Fire Department is fully staffed, trained, and equipped for emergency response.
SNF Project personnel are trained on the expected actions to be taken in case of a fire. Personnel are to notify the Hanford Fire Department, evacuate the facility, and follow approved fire response plans specific to the facility.

**D11.4.5 Fire Fighting Readiness Assurance**

A pre-fire plan for the ISA will be prepared by the Hanford Fire Department prior to facility operations. A summary of SNF Project fire prevention inspections, fire safety drills and exercises, and program record-keeping requirements is provided in Section 11.4.5 of the SNF Project FSAR.

**D11.5 REFERENCES**


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CHAPTER D12.0

PROCEDURES AND TRAINING
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D12.0 PROCEDURES AND TRAINING

D12.1 INTRODUCTION

A description of the essential features of Spent Nuclear Fuel (SNF) Project procedures and training is provided in Chapter 12.0 of the SNF Project Final Safety Analysis Report (FSAR).

D12.2 REQUIREMENTS

The requirements that form the basis for the SNF Project training and procedures programs are identified in Section 12.2 of the SNF Project FSAR.

D12.3 PROCEDURE PROGRAM

SNF Project activities are conducted in accordance with written procedures. A summary of the facility procedures program, including development and maintenance of procedures, is provided in Section 12.3 and its subsections in the SNF Project FSAR.

D12.4 TRAINING PROGRAM

The objective of the SNF Project personnel training program is to provide and maintain a qualified work force for safe and efficient facility operations. A summary of the SNF Project personnel training program—including training development, maintenance of training, and modification of training materials—is provided in Section 12.4 and its subsections in the SNF Project FSAR.
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CHAPTER D13.0

HUMAN FACTORS
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D13.0 HUMAN FACTORS

D13.1 INTRODUCTION

DOE-STD-3009-94, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports, allows for a graded approach to the application of human factors analysis to nuclear facilities. Per DOE-STD-3009-94, final safety analysis report (FSAR) discussions are to pertain only to the human-machine interfaces with safety structures, systems, and components (SSCs) and in proportion to the importance of those human-machine interfaces to the performance of the safety SSCs. Since the safety-class SSCs at the 200 Area Interim Storage Area (ISA) are all passive devices, there are no human-machine interfaces except during storage system unloading. Events associated with unloading have been analyzed in Chapter D3.0.

All equipment used for storage system handling at the ISA exists either at other U.S. Department of Energy (DOE) facilities or on the Hanford Site. The applicable procedures, training, and staffing requirements needed for 200 Area ISA operations and the existing equipment were previously defined and are currently being used at the 400 Area ISA (Fast Flux Test Facility). The storage system handling procedures and training requirements are in compliance with the DOE-RL-92-36, Hanford Site Hoisting and Rigging Manual. DOE-RL-92-36 is updated based on lessons learned both at the Hanford Site and at other DOE facilities. ISA surveillance and maintenance procedures will be prepared using the Spent Nuclear Fuel (SNF) Project Procedures Writer's Guide, which encompasses the human factors requirements of DOE-STD-1029-92, Writer's Guide for Technical Procedures. A review of the detailed operations associated with storage system handling identified no hazards that warrant analysis by a human factors expert. The approach used to address human factors is deemed adequate based on the guidance provided in DOE-STD-3009-94.

D13.2 REQUIREMENTS

The requirements that establish the basis for human factors engineering and storage system handling are identified in the following:

- DOE Order 5480.23, Nuclear Safety Analysis Reports
D13.3 HUMAN FACTORS PROCESS

The human factors process is discussed in Section D13.1.

D13.4 IDENTIFICATION OF HUMAN-MACHINE INTERFACES

The safety-class SSCs at the 200 Area ISA are all passive devices; therefore, there are no human-machine interfaces except during storage system unloading. Events associated with unloading have been analyzed in Chapter D3.0. Critical lift procedures for cask handling activities will incorporate controls for the lifting restrictions identified in Chapter D5.0.

D13.5 OPTIMIZATION OF HUMAN-MACHINE INTERFACES

See Sections D13.1 and D13.4.

D13.6 REFERENCES


CHAPTER D14.0

QUALITY ASSURANCE
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D14.0 QUALITY ASSURANCE

D14.1 INTRODUCTION

An introduction to the quality assurance program that includes the objectives and scope that apply to all Spent Nuclear Fuel (SNF) Project quality assurance activities is provided in Chapter 14.0 of the SNF Project Final Safety Analysis Report (FSAR).

D14.2 REQUIREMENTS

The requirements that form the basis of the quality assurance program are identified in Section 14.2 of the SNF Project FSAR. Additional requirements for the 200 Area Interim Storage Area (ISA) are identified in the following paragraphs. These requirements ensure the quality assurance requirements for federal repository acceptance of SNF and U.S. Nuclear Regulatory Commission (NRC) equivalency requirements are satisfied. Tables D4-1 and D4-2 present listings of safety-class and safety-significant structures, systems, and components required for the 200 Area ISA.

The U.S. Department of Energy, Richland Operations Office (DOE-RL), has directed (Sellers 1995) that DOE/RW-0333P, Quality Assurance Requirements and Description (QARD), published by the Office of Civilian Radioactive Waste Management (OCRWM), be applied as the principal quality assurance document to the SNF Project OCRWM program. DOE-RL has directed application of DOE/RW-0333P to the following SNF-related activities as they relate to repository storage:

- Characterization or data collection for input and use
- Conditioning into final form
- Handling, packaging, and transportation.

Items, activities, and documentation determined to be important to safety are presented in Table 3-1 and Section 5.0 of HNF-SD-SNF-RPT-007, Application of the Office of Civilian Radioactive Waste Management Quality Assurance Requirements to the Hanford Spent Nuclear Fuel Project. This document identifies structures, systems, components, and activities that require application of DOE/RW-0333P requirements to ensure compliance with the DOE-RL
direction. Assumptions described in Section 2.2 of HNF-SD-SNP-RPT-007, applicable to the ISA and QARD compliance, include the following:

- **324 Building Light Water Reactor (LWR) SNF** will be canistered for interim storage in a manner that does not result in the need to recanister or further condition the fuel for final disposition.

- **Neutron Radiography Facility (NRF) Training, Research and Isotope Production, General Atomics (TRIGA)** casks will not be accepted by OCRWM.

- **Fast Flux Test Facility (FFTF) SNF** will be repackaged from core component containers (CCC) prior to repository acceptance.


- **Where an activity, such as data collection, and presentation have been completed outside an approved OCRWM QARD program**, the data identification, review adequacy, and usage will be validated and qualified in accordance with OCRWM Supplement III requirements.

Quantities of site-wide SNF inventories, the current storage location, and information related to each type of fuel is provided in Table 5-1 and Figure 5-1 of HNF-SD-SNF-RPT-007. Application of the QARD requirements to these fuels must address process and storage plans for each fuel type.

In addition, specific quality requirements include: (1) periodic surveillance inspections of the SNF during storage; (2) records of all inspections after acceptance of the SNF by the SNF Project; (3) record(s) of fuel damage or other nonconformances and corrective actions after acceptance of the SNF by the SNF Project; and (4) developing and maintaining OCRWM data records for the SNF inventory, as identified in formal direction from DOE-RL. Data received by the SNF Project will be considered unqualified and tracked as such until the data passes an appropriate qualification process per the QARD.

As stated in Section 5.2.2 of HNF-SD-SNP-RPT-007, the SNF Project will develop a disposition plan for Hanford Site SNF that will identify the compliance strategy for each SNF inventory. Additionally, the SNF Project will have lead responsibility for the review of guidance on final disposition requirements from the National Spent Nuclear Fuel (NSNF) Program and will provide input to the NSNF Program on the acceptability of the draft guidance.
DOE has established a regulatory policy (Grumbly 1995) that new SNF Project facilities involved in processing K Basins SNF will achieve nuclear safety equivalency with NRC-licensed facilities. An evaluation, documented in WHC-SD-SNF-DB-002, *Spent Nuclear Fuel Project Path Forward Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities*, identified requirements to establish nuclear safety equivalency that are to be met in addition to existing and applicable DOE requirements. These requirements, except those related to the design basis earthquake, are contained in HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements*. HNF-SD-SNF-DB-004, *SNF Project Seismic Design Criteria - NRC Equivalency Evaluation Report*, contains the design basis earthquake requirements. The SNF Project has self-imposed some of these requirements to the ISA.

NRC nuclear safety equivalency requirements identified in HNF-SD-SNF-DB-003 that will be applied to the ISA include the following:

- Review and approval by DOE-RL of changes to HNF-MP-599, *Project Hanford Quality Assurance Program Description*, that could be interpreted as decreasing the quality assurance program's existing commitments for the ISA (HNF-SD-SNF-DB-003, Item 16).

- Ensure the appropriate quality requirements in existing Project Hanford Management Contract procedures and instructions, as identified in WHC-SD-SNF-DB-002, remain in effect (HNF-SD-SNF-DB-003, Item 18).

Design requirements for natural phenomena hazards, other than seismic design requirements, are identified in HNF-2524, *200 East Area Interim Storage Area Preliminary Safety Evaluation Report*.

The documents cited in this chapter identify the requirements to achieve nuclear safety equivalency with NRC-licensed facilities and to meet the requirements of DOE/RW-0333P. The quality assurance program plan for SNF Project facilities provides for implementation of these requirements. A graded approach will be used for items and activities important to safety (i.e., safety-class, safety-significant, and certain general-service items and/or activities), in accordance with HNF-MP-599 and NUREG/CR-6407, *Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety*.

**D14.3 QUALITY ASSURANCE PROGRAM ORGANIZATION**

A summary of the SNF Project quality assurance program, including summaries of safety management policies and philosophies used as a basis for the program, is provided in Section 14.3 of the SNF Project FSAR.
The SNF Project organizational structure, responsibilities, authorities, and interfaces that apply to the ISA are addressed in Chapter 17.0 of the SNF Project FSAR.

D14.4 QUALITY IMPROVEMENT

Descriptions of SNF Project management programs and processes used to correct adverse conditions affecting quality at all SNF Project facilities are provided in Section 14.4 of the SNF Project FSAR.

D14.5 DOCUMENTS AND RECORDS

A description of the SNF Project document control and records management program associated with quality assurance is provided in Section 14.5 of the SNF Project FSAR.

D14.6 QUALITY ASSURANCE PERFORMANCE

An overview of the SNF Project process to ensure that the performed work meets requirements is provided in Section 14.6 and its subsections in the SNF Project FSAR. The subsections address work processes, design activities, the procurement process, program tests and inspections, management assessments, and independent assessments.

D14.7 REFERENCES


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CHAPTER D15.0

EMERGENCY PREPAREDNESS PROGRAM
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D15.0 EMERGENCY PREPAREDNESS PROGRAM

D15.1 INTRODUCTION

A description of the philosophy, objectives, and organization of the Spent Nuclear Fuel (SNF) Project emergency preparedness program for response to emergencies at the SNF Project facilities is provided in Chapter 15.0 of the SNF Project Final Safety Analysis Report (FSAR) and Chapter A15.0, Annex A. This Annex D chapter presents emergency management information specific to the 200 Area Interim Storage Area (ISA).

D15.2 REQUIREMENTS

The requirements that form the basis for the SNF Project emergency preparedness program are identified in Section 15.2 of the SNF Project FSAR.

D15.3 SCOPE OF EMERGENCY PREPAREDNESS

Potential ISA emergencies could span the spectrum of identified emergencies for SNF Project facilities, from worker injuries to general emergencies with potential public impact. The spectrum of emergencies that the ISA emergency preparedness program is designed to encompass is described in Section 15.3 of the SNF Project FSAR and in Chapter D3.0. Chapter 4.0 of HNF-2524, 200 East Area Interim Storage Area Preliminary Safety Evaluation Report, presents a discussion of the principal accidents that could occur at the ISA.

D15.4 EMERGENCY PREPAREDNESS PLANNING

SNF Project emergency preparedness planning includes identification of emergency organizations, assessment actions, notification processes, emergency facilities and equipment, protective actions, access control, training, drills, exercises, and recovery actions. A summary of the emergency response organization that is activated for ISA emergencies is provided in Section 15.4 of the SNF Project FSAR and Section A15.4, Annex A. ISA emergencies and responses will be covered in the Canister Storage Building (CSB) building emergency plan and emergency response procedures.

The provisions in Section 15.4.1.2 of the SNF Project FSAR are fulfilled by the building emergency director (BED) at the CSB for the ISA until an Incident Command Post is established. The shift manager is the BED for both on-site hazardous and nonhazardous facilities, as stated in Section 15.4.1.2 of the SNF Project FSAR.
**D15.4.1 Emergency Response Organization**

Section 15.4.1 of the SNF Project FSAR presents information related to the organizational structure, building emergency organization, Incident Command Post, Emergency Operations Center, and support resources to meet emergency event requirements. This information also applies to the ISA.

The CSB Emergency Response Organization is responsible for establishing an Incident Command Post to respond to events occurring at the ISA facility (Section 15.4.1.2 of the SNF Project FSAR). Event response information is conveyed to and from the U.S. Department of Energy, Richland Operations Office (DOE-RL) Emergency Operations Center. DOE-RL maintains responsibility for communication with off-site agencies, as indicated in Figure 15-2 in the SNF Project FSAR. Each hazardous SNF Project facility has an emergency staff of individuals who assist in the protection of personnel, property, and the environment. Initial direction and control of an emergency response at the ISA is the responsibility of the BED prior to establishment of the Incident Command Post. Key Emergency Response Organization positions and responsibilities are discussed in the subsections that follow.

**D15.4.1.1 Building Emergency Director.** The BED, as described in Section 15.4.1.2 of the SNF Project FSAR and Section A15.4.1.2, is the emergency coordinator for hazardous/facility-related events (WAC 173-303) and has the authority to commit all SNF Project resources (equipment and personnel) in response to any emergency and to request supporting resources. Other responsibilities include: (1) implementing a building emergency plan; (2) assuring that the ISA Emergency Response Organization is fully staffed and trained; (3) initially assessing, categorizing, and classifying events; (4) notifying the Patrol Operations Center and applicable contractor and DOE management through the Occurrence Notification Center; (5) implementing protective actions; (6) establishing an initial ISA Incident Command Post; (7) controlling the event scene; (8) initiating mitigating activities; and (9) initiating recovery actions when directed. At the ISA, the BED is a certified CSB operations shift manager.

A listing of the primary and alternate BEDs by title, work location, and work telephone numbers is contained within the ISA building emergency plan. The BED is on the CSB or ISA premises during hazardous operations and is available through an “on-call” list 24 hours a day at all other times. Operations maintains a listing of on-call BED names, with work and home telephone numbers, at the Occurrence Notification Center.

**D15.4.1.2 Incident Command Post Staff.** The Incident Command Post staff is a group of SNF Project emergency response personnel assigned to an Incident Command Post established for an event. The Incident Commander, as supported by the BED, Hanford Fire Department, and Hanford Patrol Shift Commander, directs all emergency response efforts at the event scene.

Emergency response efforts for the Hanford Site are conducted by the Incident Commander, BED, and Hanford Patrol Shift Commander. The BED becomes a member of the
Annex D – 200 Area Interim Storage Area

Incident Command Post and functions under the direction of the Incident Commander. In this role, the BED continues to manage and direct ISA operations.

D15.4.1.3 Event Scene Staff. The event scene staff is composed of a Hanford Fire Department Operations Section Chief (assigned by the Incident Commander), trained support staff (including Health Physics and Industrial Hygiene staff), and (as required) Hanford Fire Department medical responders and Hanford Patrol. In addition, accountability aides are responsible for facilitating the implementation of protective actions (evacuation or take cover) and for facilitating the accountability of personnel after the protective actions have been implemented. Staging area managers are responsible for coordinating/conducting activities at the staging area. Personnel accountability aides assist the staging managers by ensuring that personnel and visitors are properly evacuated from designated staging areas to a safe location. The event scene staff, as directed by the Incident Command Post, supports actions requested by the Incident Commander and the BED.

D15.4.2 Assessment Actions

Provisions of Section 15.4.2 of the SNF Project FSAR cover hazards survey, hazards assessment, emergency action levels, consequence assessment, and monitoring activities. These provisions apply to the ISA.

An ISA emergency planning hazards assessment will be developed for the ISA for hazards that have the potential to generate an “Alert” or higher emergency. The hazards assessment will be prepared from the hazard and safety analyses that are developed and included in Chapter D3.0. The hazards assessment will also be derived from other pertinent facility documentation (e.g., safety assessment documents, interim safety basis documents, and special nuclear material accountability documents). The hazards assessment provides the technical basis for the emergency management program. The scope and extent of planning and preparedness directly corresponds to the type and scope of hazards present and the potential consequences of events.

The hazards assessment identifies and characterizes the hazards relevant to potential ISA operational emergencies. This includes determination of the following:

- A broad range of initiating events
- Accident mechanisms
- Equipment or system failures
- Event indications
- Contributing events
- Source terms
- Material release characteristics
- Topography
- Environmental transport and diffusion

D15-3 January 2000
Exposure considerations
Chemical hazards.

The hazards assessment characterizes the potential consequences to workers, the public, and the environment for each postulated accident and determines the emergency planning zone (EPZ) for each facility. The assessment also determines the emergency class, protective actions, and observable indications and criteria (emergency action levels) corresponding to the range of identified accidents.

A spectrum of potential accidents ranging from minor to beyond design basis are postulated and realistically analyzed. While not every conceivable situation will be analyzed, the hazards assessments provide the framework for response to virtually any declared emergency.

The methodology, assumptions, models, and evaluation techniques used in the hazards assessments are documented in Sections 15.4.2.3 and 15.4.2.4 of the SNF Project FSAR. Results from the ISA hazards assessment are used to develop the ISA building emergency plan elements contained in the CSB building emergency plan. Hazards assessments for the ISA are reviewed annually and updated, as necessary, in accordance with Section 15.4.2.2 of the SNF Project FSAR.

D15.4.3 Notification

Notifications, in the event of an emergency event at the ISA, will be made in accordance with the provisions of Section 15.4.3 of the SNF Project FSAR in order to mitigate consequences and to protect the health and safety of workers, the public, and the environment.

D15.4.4 200 Area Interim Storage Area Emergency Facilities and Equipment

The building emergency plan for the ISA will be prepared and issued as part of the CSB building emergency plan, in accordance with HNF-IP-0263-GEN, Building Emergency Plan Guidance. A description of the facilities that will be available for coordinating ISA emergency response activities will be specified in the building emergency plan.

Emergency equipment consisting of materials and tools that may be required to measure, control, or mitigate the consequences of an emergency at the ISA is provided in Table D15-1. Detection ranges and types of instruments for radiological and nonradiological hazardous materials will be adequate for ISA emergency conditions, as determined in Section D15.4.2. The emergency planning organization will ensure that sufficient emergency equipment is available. The location of this emergency equipment will be stated in the CSB building emergency plan.
Table D15-1. Interim Storage Area Emergency Equipment.

<table>
<thead>
<tr>
<th>Type of equipment</th>
<th>Equipment capabilities</th>
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<tbody>
<tr>
<td><strong>Fixed and portable equipment</strong></td>
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<tr>
<td>Fire protection systems – Fire detection and alarm system and dry pipe automatic</td>
<td>Assists in notifying personnel, summoning the Hanford Fire Department, and in fire</td>
</tr>
<tr>
<td>sprinkler suppression system</td>
<td>suppression</td>
</tr>
<tr>
<td>Fire system pressure alarms and/or water flow alarm</td>
<td>Assists with notifying personnel of emergency conditions</td>
</tr>
<tr>
<td>Evacuation and take cover siren</td>
<td>Assists with notifying personnel of emergency conditions and, by the type of siren,</td>
</tr>
<tr>
<td></td>
<td>expected actions</td>
</tr>
<tr>
<td>Respiratory protection (SCBA)(^{(1)})</td>
<td>Protects personnel from hazardous chemicals</td>
</tr>
<tr>
<td><strong>Portable emergency equipment</strong></td>
<td></td>
</tr>
<tr>
<td>Fire extinguishers (Types A, B, and C)</td>
<td>Assists in fire suppression</td>
</tr>
<tr>
<td>Hazardous materials spill control kits (unmounted)</td>
<td>Assists with hazardous (chemical) materials stabilization and cleanup following a spill</td>
</tr>
<tr>
<td></td>
<td>or release</td>
</tr>
<tr>
<td>Command post equipment: emergency procedures, checklists (maps and photographs of</td>
<td>Provides area and site-specific emergency information</td>
</tr>
<tr>
<td>facilities optional)</td>
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</tr>
<tr>
<td>Operational event scene equipment: radiological response vehicle, emergency</td>
<td>Assists in controlling and mitigating the event</td>
</tr>
<tr>
<td>procedures, duty cards, checklists, maps, photographs of facilities</td>
<td></td>
</tr>
<tr>
<td><strong>Protective clothing and equipment</strong></td>
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</tr>
<tr>
<td>Anti-C clothing and personal protective equipment</td>
<td>Provides contamination control (anti-C clothing for radiological and acid gear for any</td>
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<tr>
<td></td>
<td>corrosive chemicals)</td>
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<tr>
<td>Miscellaneous respiratory equipment</td>
<td>Provides respiratory protection for radionuclides; this type of respirator equipment is</td>
</tr>
<tr>
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<td>not considered to be emergency equipment</td>
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</table>

\(^{(1)}\) SCBA respirators for emergency use will be thoroughly inspected at least once a month and after each use. Records of inspection dates and findings will be maintained.

SCBA = self-contained breathing apparatus.
D15.4.5 Protective Actions

Protective actions are those actions taken to preclude or reduce the exposure of individuals or the environment impacted by hazards or unsafe conditions during an emergency event at the ISA. These protective actions are presented in Section 15.4.5 of the SNF Project FSAR and are applicable to the ISA. Protective actions for the ISA will reflect use of the emergency response planning guidelines identified in Section 15.4.2.3 of the SNF Project FSAR. The planning guidelines published in the Emergency Response Planning Guidelines (AIHA 1988) will be used during an ISA emergency response to determine protective actions for unique exposures to chemical releases (see Table 15-4 of the SNF Project FSAR). The protective action guides are also used during an emergency response to determine protective actions for unique exposures to radiological releases (see Table 15-1 of the SNF Project FSAR). DOE-RL directs the use of the published protective action guides adopted by the states of Washington and Oregon (EPA-100) in DOE/RL-94-02, Hanford Emergency Management Plan.

The Hanford Site emergency management program uses the EPZ concept to focus emergency planning activities. EPZs are designated areas where protective actions may be required. The size of a zone is determined primarily by the expected dispersion distance of a particular concentration of a substance. The two exposure pathways for both radiological and nonradiological hazardous materials are the plume exposure pathway and the ingestion exposure pathway. A description of the exposure pathways is provided in Section 15.4.5 of the SNF Project FSAR. Figure D1-1 indicates the location of the CSB. Figure 2-1 in HNF-2524 shows the location of the ISA relative to the CSB.

The plume exposure pathway EPZ is the probable area of exposure to a passing cloud (or plume) of the substance, potentially resulting in direct contact with the substance through the exterior of the body or through inhalation. The plume exposure pathway EPZ includes the area where emergency planning is conducted (1) to ensure that prompt and effective actions are taken in the event of an emergency, (2) to protect on-site personnel, and (3) to ensure public health and safety. The plume exposure pathway for the CSB (10 mi) is shown in Figure 15-6 in the SNF Project FSAR.

The ingestion exposure pathway EPZ is the probable area of exposure to contaminated foodstuffs or water potentially resulting in deposition of the material in various internal organs following ingestion (eating or drinking). The ingestion exposure pathway EPZ for radiological and nonradiological incidents at all Hanford Site facilities corresponds to the 80-km (50-mi) EPZ for Energy Northwest's (formerly known as Washington Public Power Supply System's) Nuclear Plant 2. The gray area in Figure 15-6 in the SNF Project FSAR represents the ingestion EPZ for the Hanford Site.

The protective actions required to minimize the exposure of workers and the public are summarized in Section 15.4.5 of the SNF Project FSAR. Examples of protective actions as a function of accident category and consequences are illustrated in Table 15-5 in the SNF Project FSAR.
D15.4.6 Training and Exercises

The CSB emergency organization will be formed, trained, and tested for potential ISA emergency events in accordance with the provisions of Section 15.4.6 of the SNF Project FSAR. Drills and exercises will be developed in accordance with Section 15.4.6 of the SNF Project FSAR, with sufficient scope and detail to emphasize the facility-specific emergency events and response actions applicable to the ISA.

D15.4.7 Reentry and Recovery

The provisions applicable to a ISA emergency event termination, facility entry, transition from an emergency organization to a recovery organization, and the recovery process are provided in Section 15.4.7 of the SNF Project FSAR.

D15.5 DOCUMENT CONTROL

The ISA building emergency plan, implementing procedures, reports of drills and exercises, and emergency event documentation will be controlled and updated in accordance with the provisions of Section 15.5 of the SNF Project FSAR.

D15.6 REFERENCES


CHAPTER D16.0

PROVISIONS FOR DECONTAMINATION AND DECOMMISSIONING
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D16.0 PROVISIONS FOR DECONTAMINATION AND DECOMMISSIONING

D16.1 INTRODUCTION

The provisions that apply to future decontamination and decommissioning (D&D) of Spent Nuclear Fuel (SNF) Project facilities are addressed in Chapter 16.0 of the SNF Project Final Safety Analysis Report (FSAR). Provisions specific to the 200 Area Interim Storage Area (ISA) are addressed in this Chapter D16.0.

The 200 Area ISA has been given the designation of a Hazard Category 2 facility because of the inventory contained in the cask storage systems. The facility will not experience significant, if any, contamination during interim storage of the SNF. Therefore, decontamination efforts at the end of facility life will only be required in the event that an unanticipated release(s) occurs during the life of the facility. Decommissioning efforts will also be minimal based on the simplicity of the facility design (e.g., concrete slabs, graveled storage and access areas, a fence, and lighting fixtures). The end state of the 200 Area ISA is not known at this time, but several options, ranging from facility reuse to complete dismantlement and removal, are available.

D16.2 REQUIREMENTS

The requirements that form the basis for the D&D program are found in Section 16.2 of the SNF Project FSAR. Specific requirements applicable to this chapter include the following:


D16.3 DESCRIPTION OF CONCEPTUAL PLANS

General D&D considerations applicable to all SNF Project facilities are provided in Section 16.3 of the SNF Project FSAR.
D16.3.1 Design Features

Each 200 Area ISA cask storage configuration provides confinement of radiological materials during transport of the SNF to the 200 Area ISA and during storage at the ISA. The cask storage configuration design features that are important to D&D will be controlled through the design change control process to ensure that changes to the facility will provide equal or greater consideration to D&D. Further description of the confinement design is provided in Chapter D2.0.

D16.3.1.1 Safety Features. Chapters D3.0, D4.0, D5.0, and D6.0 identify the selected safety features for preventing or mitigating the postulated accidents. These implemented engineered barriers and administrative programs will prevent or greatly reduce the consequences of any postulated upset event or accident; thus, the amount and location of any radiological material releases will be diminished for D&D cleanup activities. No design basis accident will result in off-site dose consequences (prevented) or on-site dose consequences exceeding the evaluation guidelines (mitigated).

D16.3.2 Operational Considerations

The potential for personnel or equipment contamination is minimized by the design of the cask storage systems, and by administrative controls, radiological practices, and work guidelines defined in operating procedures and work permits. Because baseline operations assume no spread of contamination from the cask storage systems, no special facilities for the support of decontamination activities have been provided. Although the risk of contamination is minimal, operating procedures address requirements for radiological control surveys in the facility. For example, surveys will be performed annually to verify that each cask system remains free of contamination.

D16.3.3 Decommissioning

The 200 Area ISA components may be removed at a future time in compliance with applicable regulations. The process used to develop the 200 Area ISA D&D plan is provided in Section 16.3.3 of the SNF Project FSAR. Conceptual plans for D&D will include an updated facility hazard analysis for the D&D activities, which will be used to prepare the plan to administer the expected D&D strategy. Conceptual plans will also include a preliminary deactivation plan that contains at least the following information:

- Structures, systems, and components in their final configuration
- Review and determination of the status of structures, systems, and components based on the mission and life-cycle phase
Configuration management for missing or inaccurate design baseline documentation, voiding and downgrading of design documents, and turnover of design baseline documents to the environmental restoration contractor.

Decommissioning plans for the 200 Area ISA facility will be developed and reviewed against the existing environmental impact statement. The environmental impact statement will be updated to include D&D activities in accordance with the *National Environmental Policy Act* process, if deemed appropriate.

The construction of the 200 Area ISA as an above-grade facility will simplify decontamination and dismantling. The ISA storage building will be decontaminated as necessary and dismantled using conventional techniques. The ISA concrete pads will be dismantled and removed from the site.

**D16.4 REFERENCES**


*National Environmental Policy Act (NEPA) of 1969*, 42 USC, 4321, et seq.
CHAPTER D17.0
MANAGEMENT, ORGANIZATION, AND INSTITUTIONAL SAFETY PROVISIONS
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D17.0 MANAGEMENT, ORGANIZATION, AND INSTITUTIONAL SAFETY PROVISIONS

D17.1 INTRODUCTION

A description of the organizational structure, responsibilities, and interfaces that support safe design, construction, and operational activities of the 200 Area Interim Storage Area (ISA), as a subproject of the Spent Nuclear Fuel (SNF) Project, are addressed in Chapter 17.0 of the SNF Project Final Safety Analysis Report (FSAR).

D17.2 REQUIREMENTS

The requirements that form the basis for management, organizational, and safety provisions are identified in Section 17.2 of the SNF Project FSAR.

D17.3 ORGANIZATIONAL STRUCTURE, RESPONSIBILITIES, AND INTERFACES

The overall organizational structure, responsibilities, and interfaces for ISA operations are identified in Section 17.3 of the SNF Project FSAR.

D17.4 SAFETY MANAGEMENT POLICIES AND PROGRAMS

The safety management policies and programs applicable to ISA are identified in Section 17.4 of the SNF Project FSAR.
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7. Purchase Order No.: 
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8. Originator Remarks: 
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  - 3. Information
  - 4. Review
  - 5. Post-Review
  - 6. Dist. (Receipt Acknow. Required)

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(See Approval Designator for required signatures)

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Spent Nuclear Fuel Project
Canister Storage Building
Final Safety Analysis Report

Prepared for the U.S. Department of Energy
Assistant Secretary for Environmental Management

Project Hanford Management Contractor for the
U.S. Department of Energy under Contract DE-AC06-96RL13200

Fluor Hanford
P.O. Box 1000
Richland, Washington

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HNF-3553 Rev 0 Annex A

#### C. Title

Spent Nuclear Fuel Project Final Safety Analysis Report Annex A Canister Storage Building

#### E. Required Information

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2. Internal Review Required? **Yes**

3. References in the Information are Applied Technology? **Yes**

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6. Will Material be Handed Out? **Yes**

#### H. Author/Requestor

I. J. Garvin (Print and Sign)  
R. E. Boyleston (Print and Sign)

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### Additional Comments

If Additional Comments, Please Attach Separate Sheet

**Information Clearance Approval**

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Spent Nuclear Fuel Project Canister Storage Building Final Safety Analysis Report

L. J. Garvin
Fluor Hanford

Date Published
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Prepared for the U.S. Department of Energy
Assistant Secretary for Environmental Management

Project Hanford Management Contractor for the
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Fluor Hanford
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EXECUTIVE SUMMARY

AE.1 FACILITY BACKGROUND AND MISSION

The U.S. Department of Energy (DOE) established the Spent Nuclear Fuel (SNF) Project to address safety and environmental concerns associated with deteriorating SNF presently stored under water in the Hanford Site K Basins, which are located in the 100 K Area near the Columbia River. Recommendations for a series of projects to construct and operate systems and facilities to manage the safe removal of K Basins fuel were made in WHC-EP-0830,汉福尔德退役核燃料项目建议路径前进，1 and its subsequent update, WHC-SD-SNF-SP-005，汉福尔德退役核燃料项目整合流程策略为K屏蔽燃料。2 The integrated process strategy recommendations include the following steps:

- Fuel preparation activities at the K Basins, including removing the fuel elements from their K Basins canisters, separating fuel particulate from fuel elements and fuel fragments greater than 0.25 in. in any dimension, removing excess sludge from the fuel fragments by means of flushing, as necessary, and packaging the fuel into multi-canister overpacks (MCOs)

- Transportation of MCOs loaded with SNF from K Basins to the Cold Vacuum Drying Facility (CVDF)

- Removal of free water by draining and vacuum drying at the CVDF in the 100 K Area

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Dry shipment of fuel from the CVDF to the Canister Storage Building (CSB), a new facility in the 200 East Area.

Interim storage of the MCOs in the CSB until a suitable long-term repository is established.

The purpose of this final safety analysis report (FSAR) is to provide the basis for authorization to proceed with the operation of the CSB. The scope of this report includes operating and support structures and equipment required for receipt, handling, and interim storage of the MCOs.

The SNF Project FSAR is a multi-volume document. Volume 1 contains SNF Project information applicable to new SNF Project facilities. This Volume 2, facility FSAR Annex A, contains information specific to the CSB. The information contained in this Volume 2, facility FSAR Annex A, is based on the CSB design information. All topics recommended in DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports,* are addressed, including descriptions of site and facility design, hazard and accident analyses, safety-class and safety-significant equipment, derivation of technical safety requirements (TSRs), prevention of inadvertent criticality, and other areas of facility design and operational programs. This report also is based on implementation of U.S. Nuclear Regulatory Commission nuclear safety equivalency with the additional requirements compiled and approved in HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward Additional NRC Requirements.* Future changes in facility mission are anticipated to require revisions to Volume 2, CSB FSAR Annex A. These changes include the interim storage of other fuel types (e.g., fuel from the Shippingport pressurized water reactor) and the storage of glass canisters.

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from the Hanford Waste Vitrification Plant. These future missions are not expected to impact the
current hazard classification as a Category 2 facility.

AE.2 FACILITY OVERVIEW

The CSB facility provides for receiving, sampling, monitoring, and interim storage of
MCOs. The partially constructed foundation of the canceled Hanford Waste Vitrification Plant
has been incorporated into the design of the CSB, with some modifications made to meet the
different SNF Project mission needs.

The CSB facility is a steel-framed building that encloses the operating area; the
load-in/load-out area; and three equally sized, below-grade concrete vaults. The concrete vaults
are covered by a concrete operating deck. Support functions and equipment are housed in a
steel-framed support area building adjacent to the north side of the operating area building. Only
the northernmost vault (vault 1) in the operating area is equipped with steel tubes for staging
mechanically sealed MCOs and for interim storage of the MCOs with welded caps. For
completeness, vaults 2 and 3 will be mentioned or described in this document; however,
operations in vaults 2 and 3 are outside the scope of this FSAR.

An MCO received at the CSB contains five or six baskets of dried SNF that are stacked
one on top of the other. The stainless steel outer shell and bottom of the MCO and the
mechanically-sealed top carbon steel shield plug are designed to meet the *Boiler and Pressure
Vessel Code* and to provide effective long-term confinement of the enclosed SNF. The shield
plug also provides access to the interior of the MCO through four access ports on the top of the
shield plug assembly. The access ports provide the means for sampling those MCOs that are to be
monitored at the CSB. The four access ports will be sealed at the CVDF using gasketed, bolted,

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Engineers, New York, New York.
cover plates before the MCO is transported to the CSB. At the load-in/load-out area, a receiving crane is used to lift the transportation cask that contains the MCO off the transportation trailer in the vestibule area and then transfer the cask–MCO to the cask receiving pit. The CSB MCO handling machine (MHM) is used to lift the MCO out of its transportation cask and transport the MCO across the operating deck to the CSB sampling/weld area or down into a storage tube in vault 1.

The CSB operating deck is a 5-ft-thick, at-grade, reinforced concrete structure. The dimensions of the operating deck are approximately 230 ft (north–south) by 137 ft (east–west). The operating area deck floor is bounded to the north by the load-in/load-out area (trailer vestibule and MCO service station) and support area building foundations, and to the south by the sampling/weld area foundation. The operating deck structure contains numerous full-thickness, embedded steel sleeves that receive the storage tubes and tube plugs for standard storage tubes and overpack storage tubes in vault 1. The embeds also provide a location for securing the tube plug cover plates in vault 1 and the deck embed cover plates in vaults 2 and 3. These embeds are offset and arranged at a center-to-center distance of 4 ft, 8 in. east–west and 4 ft, 6 in. north–south. The distance between MCOs has been evaluated for prevention of inadvertent criticality and has been found to be adequate.

The sampling/weld area, located on the south end of the operating area, contains equipment for pressure-checking and sampling the MCOs and welding equipment for installing cover caps on the MCOs.

Vault 1 contains 220 standard storage tubes and 6 overpack storage tubes. Vault 1 is cooled by natural convection through an above-grade inlet structure and exhaust stack, through below-grade concrete intake and exhaust plenums. The CSB subsurface structure provides shielding for the intake and exhaust plenums. The 3-ft-thick interior walls of the subsurface structure provide shielding from the source term associated with SNF storage. The exterior walls were designed to meet the shielding criteria given in Title 10, Code of Federal Regulations,
Part 835, "Occupational Radiation Protection" (10 CFR 835),\(^6\) for uncontrolled access areas. Assuming that each MCO contains the maximum activity value associated with Mark IV fuel, shielding calculations indicate a dose rate at the base of the intake stack of less than 0.05 mrem/hour. Heating of concrete by the radiation field inside the vault has been calculated to be 6.3 x \(10^4\) W. This value has been determined to be insignificant compared with convective heat transfer of 161.2 kW (CSB-HV-0001\(^7\)).

Each standard storage tube is capable of staging or storing two MCOs. Each overpack storage tube is available to accommodate an abnormal or suspect MCO, as described in an associated recovery plan. The vault 1 storage tubes are supported from the foundation base slab of the vault and are accessed through penetrations in the operating deck. These embedments or storage tubes are normally closed with removable tube plugs. The standard tube plugs provide radiation shielding, a filtered vent, and connections for sampling the storage tube atmosphere. The overpack storage tube plugs have connections for sampling, purging, and pressure-relief of the storage tube atmosphere; pressure gauges for surveillance of the tube pressure; and lock-down devices. Each standard storage tube contains both a bottom impact absorber and an intermediate impact absorber to mitigate the impact of dropped MCOs. Each overpack storage tube is fitted with a bottom impact absorber.

The figures in Chapter A1.0 show the CSB site plan and those in Chapter A2.0 show the exterior view of the CSB structure, building floor plan and sectional views, an MCO assembly, and a general arrangement of the MHM.

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AE.3 FACILITY HAZARD CLASSIFICATION

A final hazard categorization of the CSB facility was performed based on the final hazard and accident analyses. The categorization estimated the amount of radiological material at risk for the hazard categorization in accordance with DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*. Consistent with DOE-STD-1027-92, the final categorization was based on the unmitigated release of available radiological inventories. These CSB material quantities were compared against the threshold quantities contained in DOE-STD-1027-92. The final categorization of the CSB was determined to be Hazard Category 2.

AE.4 SAFETY ANALYSIS OVERVIEW

The scope of the hazard analysis included all normal, intended CSB operations for receiving, handling, and storing sealed MCOs. The analysis included reviews of operations flow diagrams and current operating procedures. CSB operations considered were those involved with bringing the transportation cask containing an MCO into the facility on the transporter, moving the cask-MCO to the load-in/load-out area and removing the cask lid, transporting the MCO from the load-in/load-out area to either the sampling/weld area or the storage tube with the MHM, conducting activities during MCO staging and interim storage, transporting the MCO from the storage tube to the sampling/weld area, and returning it from the sampling/weld area after sampling or welding.

The hazard analysis involved the identification of hazard sources, hazardous conditions, potential accident scenarios and their initiators, and preliminary assessments of event frequencies.

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and consequences. Hazards were identified by type and quantity, and represent a complete spectrum of events that could occur throughout the facility.

Bounding and representative accidents were selected from the hazard analysis for further quantitative evaluation as design basis accidents (DBAs). These accidents are presented in detail in Chapter A3.0, Section 3.4. Each DBA analyzed represents a bounding case for a category of hazards and accidents. The six major DBA events evaluated are:

- Mechanical damage of the MCO (Section A3.4.2.1)
- Gaseous release from the MCO (Section A3.4.2.2)
- MCO internal hydrogen deflagration (Section A3.4.2.3)
- MCO external hydrogen deflagration (Section A3.4.2.4)
- Thermal runaway reactions inside the MCO (Section A3.4.2.5)
- Violation of design temperature criteria (Section A3.4.2.6).

Three receptors were evaluated in the DBA analysis:

- Hanford Site boundary receptor (offsite), defined at a distance of 17,390 m east of the CSB
- Collocated worker receptor (onsite), defined at a distance of 100 m east of the CSB
- Highway 240 receptor, defined at a distance of 9,280 m west of the CSB.

Consequences to the Hanford Site boundary receptor and collocated worker receptor were compared against defined release limits and evaluation guidelines and were used for the selection of safety-class and safety-significant features, respectively. Lacking any defined evaluation guideline for the Highway 240 receptor group, consequences to those receptors were calculated...
for informational purposes only. In addition, the hazard analysis identified required worker safety protection features on a semi-quantitative basis.

The safety-class features relied upon in the facility safety basis are described and tabulated in Chapter A4.0, Section A4.3 and are listed as follows:

- CSB subsurface structures, including vaults, air intake, and exhaust plenums
- Carbon steel base slab embeds for the standard and overpack storage tubes
- CSB at-grade structures, including operating deck (including the sampling/weld area and load-in/load-out area), bases for intake tower and exhaust stack, and support area building foundation
- Standard storage tubes, bottom impact absorbers, and tube base assemblies
- Overpack storage tubes and tube base assemblies
- CSB intake structure and exhaust stack
- MCOs
- Transportation cask
- MHM seismic restraint system
- MHM rails and rail frogs.
The safety-significant features relied upon in the facility safety basis are described and tabulated in Chapter A4.0, Section A4.4 and are listed as follows:

- Operating area shelter and support area building
- Standard storage tube plugs
- Overpack storage tube plugs
- MHM structural components, MCO grapple and hoist, and MHM interlocks P2, P6, and P21
- Tube vent and purge cart
- Receiving crane structure and hoist
- Cask receiving pit
- Transportation cask servicing system
- MCO sampling system
- Standard storage tube intermediate impact absorber
- MHM fixed shielding
- Transportation cask shielding
- MCO shield plug shielding
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- Standard storage tube lower flange, bottom impact absorber, and standard interface guide ring funnel

- Overpack storage tube lower flange, bottom impact absorber, and overpack interface guide ring funnel

- Cask lifting yoke

- MCO centering guide

- Cask receiving impact absorber

- Sampling/weld station impact absorber and shield halves

- Shield hatch and MCO guide assembly

- Helium rupture disk.

AE.5 ORGANIZATIONS

Fluor Hanford, Incorporated, is responsible to the DOE for the planning, integration, and management of SNF Project activities, including programs, projects, and operations. Organizational responsibilities related to the SNF Project are summarized in Volume 1, "Executive Summary," and described in detail in Chapter 17.0 of that volume.

a-csbes sur AES-10 March 2000
AE.6 SAFETY ANALYSIS CONCLUSIONS

The safety analysis conclusions for the CSB DBAs are presented in detail in Chapter A3.0, Section A3.4.2 and are summarized in the following paragraphs.

 Mechanical Damage of Multi-Canister Overpack. An MCO, or the cask–MCO combination, subjected to an accidental drop, impact, or shear of sufficient magnitude could be damaged such that the MCO is breached or its internal geometry is compromised. Several potential accidents at the CSB that could damage the MCO with mechanical forces were identified in the hazard analysis. Three classifications of accidents were evaluated: drop of the MCO or cask–MCO, shear of the MCO by the MHM, and impacts to the MCO other than drops and shears. The unmitigated radiological offsite doses for all mechanical damage events are below the offsite release limits while the onsite doses are within evaluation guidelines for unlikely events. For safety features selected to protect analysis assumptions or for defense-in-depth structures, systems, and components (SSCs) that help prevent or mitigate other events within this accident category, see Section A3.4.2.

 Gaseous Release from the Multi-Canister Overpack. Events that may result in releases of radioactive material to the CSB facility from the uncontrolled release of the MCO’s internal gas pressure were evaluated. A pressurized gaseous release would lead to entrainment and release of fuel particulate from the MCO and the creation of a radiological hazard. Several potential accidents at the CSB that could lead to a gaseous release were identified in the hazard analysis, including overpressurization caused by radiolytic decomposition and gaseous release caused by sampling system failure or operator error during sampling and backfilling operations at the CSB. The unmitigated radiological offsite doses from all credible gaseous release events were calculated to be below the offsite release limits while the onsite doses were within evaluation guidelines for events in the anticipated category. For safety features selected to protect analysis assumptions or for defense-in-depth SSCs that help prevent or mitigate other events within this accident category, see Section A3.4.2.
Multi-Canister Overpack Internal Hydrogen Deflagration. Events were evaluated that could lead to the formation of flammable mixtures of hydrogen and oxygen within an MCO, which if ignited, could result in a deflagration inside the MCO. Several potential accidents at the CSB that could lead to an internal hydrogen deflagration were identified in the hazard analysis, including radiolytic decomposition of oxygen-containing compounds, introduction of oxygen into the MCO during recharging at the sampling/weld station, and ingress of oxygen following an MCO breach. The unmitigated radiological offsite doses from all internal hydrogen deflagration events were calculated to be below the offsite release limits while the onsite radiological doses were within evaluation guidelines for events in the unlikely category. For safety features selected to protect analysis assumptions or for defense-in-depth SSCs that help prevent or mitigate other events within this accident category, see Section A3.4.2.

Multi-Canister Overpack External Hydrogen Deflagration. Events were evaluated that could lead to the release of hydrogen from the MCO. The uncontrolled release of hydrogen could result in the formation of flammable mixtures of hydrogen and oxygen outside an MCO, which if ignited, could result in a deflagration. Several potential scenarios at the CSB that could lead to an external deflagration were identified in the hazard analysis, including deflagrations at the sampling/weld station, MHM, and storage tube. The unmitigated radiological offsite doses from all credible external hydrogen deflagrations events were calculated to be below the offsite release limits while the onsite doses were within evaluation guidelines for events in the anticipated category. For safety features selected to protect analysis assumptions or for defense-in-depth SSCs that help prevent or mitigate other events within this accident category, see Section A3.4.2.

Thermal Runaway Reactions inside the Multi-Canister Overpack. A thermal runaway reaction is only possible in an MCO containing extremely high temperature fuel and excessive amounts of water. The accident evaluations for this DBA demonstrate that a thermal runaway accident is not credible at the CSB provided the fuel is properly loaded into the MCO at the K Basins and the dryness tests at the CVDF are satisfied. No release results from this event and the offsite release limits and onsite evaluation guidelines are satisfied. For safety features selected...
to protect analysis assumptions or for defense-in-depth SSCs that help prevent or mitigate other
events within this accident category, see Section A3.4.2.

Violation of Design Temperature Criteria. The hazard analysis identified the scenario for
this accident as one in which the MCO and the safety-class CSB concrete structures could exceed
their design temperatures due to a lack of cooling. The MCOs and CSB have been designed to
provide sufficient heat transfer from the MCO so that unacceptably high temperatures will not be
reached during normal handling and storage of the MCO at the CSB. Evaluation of the DBA
demonstrates that situations in which a reduction in normal heat conduction causes overheating
are precluded by the design of facility features such as the CSB vault and intake and exhaust
structures. The accident is prevented and offsite release limits and onsite evaluation guidelines are
satisfied. For safety features selected to protect analysis assumptions or for defense-in-depth
SSCs that help prevent or mitigate other events within this accident category, see Section A3.4.2.

AE.7 SAFETY ANALYSIS REPORT ORGANIZATION

The purpose of this section of the Executive Summary as outlined in the guidance from
DOE-STD-3009-94 is to describe the structure and content of the document, particularly for
cases in which the Standard was not followed. This report is based on the format and content
guidance of DOE-STD-3009-94 and the requirements of DOE Order 5480.23, Nuclear Safety
Analysis Reports. This report also includes selected content guidance from Title 10, Code of
Federal Regulations, Part 72, "Licensing Requirements for the Independent Storage of Spent
Nuclear Fuel and High-Level Radioactive Waste," Section 72.24, "Contents of Application:
Technical Information," and NRC Regulatory Guide 3.48, Standard Format and Content for the

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9DOE Order 5480.23, Nuclear Safety Analysis Reports, U.S. Department of Energy,
Washington, D.C.

1010 CFR 72, 1995, "Licensing Requirements for the Independent Storage of Spent
Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage),\textsuperscript{11} as a result of the DOE regulatory policy implemented by HNF-SD-SNF-DB-003.\textsuperscript{4}

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<tr>
<td>AC</td>
<td>Administrative Control</td>
</tr>
<tr>
<td>ALARA</td>
<td>as low as reasonably achievable</td>
</tr>
<tr>
<td>ARF</td>
<td>airborne release fraction</td>
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<tr>
<td>ARR</td>
<td>airborne release rate</td>
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<tr>
<td>BDBA</td>
<td>beyond design basis accident</td>
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<tr>
<td>BED</td>
<td>building emergency director</td>
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<tr>
<td>BR</td>
<td>breathing rate</td>
</tr>
<tr>
<td>CAEM</td>
<td>continuous airborne effluent monitor</td>
</tr>
<tr>
<td>CAM</td>
<td>continuous air monitor</td>
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<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
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<tr>
<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
</tr>
<tr>
<td>D&amp;D</td>
<td>decontamination and decommissioning</td>
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<tr>
<td>DBA</td>
<td>design basis accident</td>
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<td>DBE</td>
<td>design basis earthquake</td>
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<tr>
<td>DCS</td>
<td>distributed control system</td>
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<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
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<tr>
<td>EF</td>
<td>error factor</td>
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<tr>
<td>EOC</td>
<td>Emergency Operations Center</td>
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<td>EPZ</td>
<td>emergency planning zone</td>
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<tr>
<td>ERO</td>
<td>emergency response organization</td>
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<td>ERPG</td>
<td>Emergency Response Planning Guideline</td>
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<td>FFF</td>
<td>Fast Flux Test Facility</td>
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<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
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<tr>
<td>HEPA</td>
<td>high-efficiency particulate air (filter)</td>
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<tr>
<td>HFE</td>
<td>human factors engineering</td>
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<tr>
<td>HMI</td>
<td>human-machine interface</td>
</tr>
<tr>
<td>HPT</td>
<td>health physics technician</td>
</tr>
<tr>
<td>HVAC</td>
<td>heating, ventilation, and air conditioning</td>
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<tr>
<td>HWVP</td>
<td>Hanford Waste Vitrification Plant</td>
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<tr>
<td>IFMSF</td>
<td>Irradiated Fissile Material Storage Facility</td>
</tr>
<tr>
<td>ISA</td>
<td>Interim Storage Area</td>
</tr>
<tr>
<td>ITS</td>
<td>important to safety</td>
</tr>
<tr>
<td>$k_{\text{eff}}$</td>
<td>effective neutron multiplication factor</td>
</tr>
<tr>
<td>LCO</td>
<td>Limiting Condition for Operation</td>
</tr>
<tr>
<td>LPF</td>
<td>leak path factor</td>
</tr>
<tr>
<td>MAR</td>
<td>material at risk</td>
</tr>
<tr>
<td>MCC</td>
<td>motor control center</td>
</tr>
<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
</tr>
<tr>
<td>MHM</td>
<td>multi-canister overpack handling machine</td>
</tr>
<tr>
<td>MLRS</td>
<td>Multiple Launch Rocket System</td>
</tr>
<tr>
<td>MTU</td>
<td>metric ton of uranium</td>
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</table>
NA not applicable
NPH natural phenomena hazard
NRC U.S. Nuclear Regulatory Commission
OCRWM Office of Civilian Radioactive Waste Management
PFP Plutonium Finishing Plant
PMP probable maximum precipitation
PUREX Plutonium-Uranium Extraction (Facility)
QARD Quality Assurance Requirements and Description (document)
RALS roller arm limit switch
RF respirable fraction
RH relative humidity
RL U.S. Department of Energy, Richland Operations Office
RPP River Protection Project
SNF spent nuclear fuel
SNV standard normal variable
SPSS solenoid-powered switch striker
SRSS square root of the sum of the squares
SSC structure, system, and component
SSI soil-structure interaction
TSR technical safety requirement
UD unit dose
UPS uninterruptible power supply
WESF Waste Encapsulation and Storage Facility
WNP Washington Nuclear Plant
CHAPTER A1.0

SITE CHARACTERISTICS
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<td>Canister Storage Building</td>
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<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
</tr>
<tr>
<td>ERPG</td>
<td>Emergency Response Planning Guideline</td>
</tr>
<tr>
<td>FFTF</td>
<td>Fast Flux Test Facility</td>
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<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
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<td>HWVP</td>
<td>Hanford Waste Vitrification Plant</td>
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<tr>
<td>ISA</td>
<td>Interim Storage Area</td>
</tr>
<tr>
<td>MLRS</td>
<td>Multiple Launch Rocket System</td>
</tr>
<tr>
<td>NRC</td>
<td>U.S. Nuclear Regulatory Commission</td>
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<tr>
<td>PFP</td>
<td>Plutonium Finishing Plant</td>
</tr>
<tr>
<td>PUREX</td>
<td>Plutonium-Uranium Extraction (Facility)</td>
</tr>
<tr>
<td>RPP</td>
<td>River Protection Project</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
</tr>
<tr>
<td>SSC</td>
<td>structure, system, and component</td>
</tr>
<tr>
<td>WESF</td>
<td>Waste Encapsulation and Storage Facility</td>
</tr>
<tr>
<td>WNP</td>
<td>Washington Nuclear Plant</td>
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</table>
A1.0 SITE CHARACTERISTICS

A1.1 INTRODUCTION

The Canister Storage Building (CSB) is in the 200 East Area of the U.S. Department of Energy (DOE) Hanford Site. The CSB will be used to store the spent nuclear fuel (SNF) generated by the Hanford Site plutonium production reactors and previously stored under water in the 100 Area K Basins. The objective of this chapter is to describe the characteristics of the site on which the CSB is located that support the hazard analysis and accident analyses presented in Chapter A3.0. See Chapter A2.0 for a detailed description of the CSB structure and systems. This chapter and Chapter 1.0 of the SNF Project Final Safety Analysis Report (FSAR) contain information related to regional and Hanford Site characteristics.

A1.2 REQUIREMENTS

The requirements that establish the basis for CSB siting are identified in Section 1.2 of the SNF Project FSAR. Some of the requirements listed in Section 1.2 of the SNF Project FSAR were not in existence at the time the below-grade portion of the CSB was designed. The impact of the later orders and standards is reflected in WHC-SD-SNF-DB-009, Canister Storage Building Natural Phenomena Hazards. Also see Section 1.2 of the SNF Project FSAR for discussion of U.S. Nuclear Regulatory Commission (NRC) equivalency requirements.

In addition to the requirements identified in Section 1.2 of the SNF Project FSAR, the following industry standards are applicable to the CSB safety basis:

- ASCE 7-93, Minimum Design Loads for Building and Other Structures

A1.3 SITE DESCRIPTION

The following sections address the geography, demography, and regional land and water use of the area encompassed by and surrounding the CSB site. See Section 1.3 of the SNF Project FSAR for a description of the Hanford Site and associated areas.

A1.3.1 Geography

A1.3.1.1 Hanford Site Vegetation. See Section 1.3.1.1 of the SNF Project FSAR for information applicable to all SNF Project facilities.
A1.3.1.2 Hanford Site Facilities. See Section 1.3.1.2 of the SNF Project FSAR for information applicable to all SNF Project facilities.

A1.3.1.3 Boundaries for Evaluation of Accident and Effluent Release Limits. As indicated in Section A3.1, three receptor locations were used to determine the consequences of accident releases to collocated workers and to the public (offsite receptors). These three locations are as follows:

- Hanford Site boundary (17,390 m east of the CSB) — defined release limits; used for calculation of offsite doses and selection of safety-class features (Sellers 1997, Scott 1995)

- Collocated worker (100 m from the CSB) — defined risk evaluation guidelines; used for calculation of onsite doses and selection of safety-significant features (Sellers 1997)

- Highway 240 (onsite, approximately 9,280 m west of the CSB) — no defined evaluation guideline; doses calculated for informational purposes only (Scott 1995).

Consequences of accidental releases from the CSB to collocated workers are calculated in Section A3.4.2 at 100 m from the point of release in accordance with approved procedures. Routine and accidental releases to the public (offsite receptor) are calculated at the Hanford Site boundary shown in Figure A1-1. This Hanford Site boundary is also the location of the controlled area boundary as the term is defined in Title 10, Code of Federal Regulations, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR 72), Section 106, "Controlled Area of an ISFSI or MRS." Before changes to the Site boundaries are placed into effect by the proper authorities, the calculations in this FSAR will be reviewed (and revisited if required) in accordance with DOE Order 5480.21, Unreviewed Safety Questions.

A1.3.2 Demography

See Section 1.3.2 of the SNF Project FSAR for the description of the demography surrounding the Hanford Site. Figure A1-2 shows the 200 East Area onsite employee population.

A1.4 ENVIRONMENTAL DESCRIPTION

A1.4.1 Meteorology

See Section 1.4.1 of the SNF Project FSAR for meteorology information applicable to all SNF Project facilities.
The specific air transport factors used for accident consequence analyses, and the basis for their calculation and use, are discussed in further detail in HNF-SD-SNF-TI-059, *A Discussion on the Methodology for Calculating Radiological and Toxicological Consequences for the Spent Nuclear Fuel Project at the Hanford Site*, and in Section A3.4.1.2. Data are provided for both ground-level and stack releases. To support the accident analyses of Section A3.4.2, air transport factors \(X/Q\) and \(WQ\), representing the dilution of a contaminant by atmospheric turbulence and diffusion as the contaminant travels downwind, were calculated (HNF-SD-SNF-TI-059). The symbol \(X/Q\) is the ratio of the average air concentration at the receptor to the average release rate at the release point. It is used to assess potential radiological dose and noncorrosive chemical concentration at downwind locations. The symbol \(WQ\) is the normalized peak air concentration at the center of a puff divided by the quantity released and is used to assess the consequences to a receptor for corrosive chemicals. The \(X/Q\)'s for the analyses were calculated using joint frequency distribution data so as to be exceeded only 0.5% of the time (99.5 percentile) for each sector, or to be exceeded only 5% of the time (95 percentile) for data from all sectors combined (the greater of the two calculated values is used in the analyses).

The GXQ computer code, Version 4.0, described in WHC-SD-GN-SWD-30002, *GXQ Program User's Guide*, was used to generate \(X/Q\) and \(WQ\) values. GXQ incorporates the methods described in NRC Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*. The specific air transport factors used for accident consequence analyses, and the basis for their calculation and use, are discussed in detail in HNF-SD-SNF-TI-059 and in Section A3.4.1.2. Table A3-7 provides the atmospheric transport factors used in accident analyses for the CSB. Supporting information also is provided in Section 1.4.1.2.8 of the SNF Project FSAR.

The wind rose data (Figure A1-3 and Figure A1-4) indicate that winds from the west-northwest sector occur most frequently (nearly 20% of the time). That is, the emissions are transported toward the east-southeast sector. Winds out of the northwest and west also occur with a relatively high frequency (12% and 9%, respectively). Information provided in these figures was obtained from data presented in HNF-SD-SNF-TI-059.

**A1.4.2 Hydrology**

See Section 1.4.2 of the SNF Project FSAR for Hanford Site hydrology information that is applicable to all SNF Project facilities.

**A1.4.2.1 Surface Water.** See Section 1.4.2.1 of the SNF Project FSAR for surface water information.

**A1.4.2.2 Vadose Zone.** See Section 1.4.2.2 of the SNF Project FSAR for a definition of the vadose zone.
A1.4.2.3 Aquifers. The 200 Area aquifer systems are discussed in Section 1.4.2.3 of the SNF Project FSAR.

A1.4.3 Geology

See Section 1.4.3 of the SNF Project FSAR for a description of Hanford Site geology that applies to all SNF Project facilities.

A1.4.3.1 Physiographic Setting of the Hanford Site. See Section 1.4.3.1 of the SNF Project FSAR for information on the physiographic characteristics applicable to the CSB.

A1.4.3.2 Stratigraphy. See Section 1.4.3.2 of the SNF Project FSAR for information on the stratigraphy of the Pacific Northwest and the Hanford Site that is applicable to the CSB.

A1.4.3.3 History of Cataclysmic Flooding in the Pasco Basin. See Section 1.4.3.3 of the SNF Project FSAR for information on cataclysmic floods that is applicable to the CSB.

A1.4.3.4 Geologic Structures of the Columbia Basin and Hanford Site. See Section 1.4.3.4 of the SNF Project FSAR for information on geologic structures that is applicable to the CSB.

A1.4.3.5 Geology of the 200 East Area Canister Storage Building. The topology and geology of the CSB site, located in the 200 East Area, is summarized on Figures A1-5 through A1-7. The suprabasalt sediments consist of the Ringold Formation and Hanford formation. The Ringold Formation conforms to the basalt bedrock surface and tilts southeast toward the axis of the Cold Creek syncline. The Ringold Formation is dominated by gravel units E and A, which are separated by the Lower Mud unit. These are the main unconfined aquifers. The Ringold Formation thins from 164 ft at the south end to nearly pinching out at the north end. The beds have been truncated and are unconformably overlain by the Hanford formation. The Hanford formation is mainly sands and gravelly sands that are between 246 and 328 ft thick at the site.

During excavation of the pit for the CSB, geologic mapping showed that the foundation for the building is in the fine-grained, laminated sands of the Hanford formation. No faults were found cutting the unit but clastic dikes were observed along the south wall of the site.

A1.4.3.6 Tectonic Development of the Hanford Site. See Section 1.4.3.6 of the SNF Project FSAR for information on geologic structures and faults that relate to Hanford Site seismic hazard analysis that is applicable to the CSB. See also WHC-SD-W236A-T1-002, Probabilistic Seismic Hazard Analysis, DOE Hanford Site, Washington.

A1.4.3.7 Contemporary Stress and Strain. See Section 1.4.3.7 of the SNF Project FSAR for information on earthquake activity, contemporary stress measurements, and subsidence history that is applicable to the CSB.
A1.4.3.8 Geologic Hazards.

A1.4.3.8.1 Seismic Hazard Assessment. The mean seismic hazard curves for the 200 East Area and the CSB site are shown in Figures A1-8 and A1-9 and illustrate the contributions of individual folds to the hazard. The relative contribution of crustal and Cascadia Subduction sources at the 200 East Area is illustrated in Figure A1-10 and the relative contribution of the three crustal sources for the same location is shown in Figure A1-11.

Horizontal and vertical equal-hazard response spectra were developed for the CSB site. These are shown at 5% damping for performance category 3 in Figure A1-12 (WHC-SD-SNF-DB-009). More detail and additional damping values are presented in WHC-SD-W236A-TI-002.

The CSB is using the design and completed portions of the subgrade construction of the canceled Hanford Waste Vitrification Plant (HWVP). The horizontal seismic response spectrum for the HWVP was a Newmark and Hall median spectra shape anchored at 0.35 g according to WHC-SD-SNF-DB-004, *Spent Nuclear Fuel Project Seismic Design Criteria — Nuclear Regulatory Commission Equivalency Evaluation Report*. The vertical spectrum is two-thirds of the horizontal. The horizontal and vertical Newmark and Hall response spectra envelop the horizontal and vertical equal hazards response spectra (respectively) (Figure A1-12). Therefore, use of the HWVP response spectra fulfills the requirements of performance category 3 response spectra (WHC-SD-SNF-DB-009).

A1.4.3.8.2 Volcanic Hazard Assessment. See Section 1.4.3.8.2 of the SNF Project FSAR for information about volcanic hazards applicable to all SNF Project facilities.

A1.4.3.8.3 Subsurface Stability. The CSB is constructed in flood sediments, the youngest sediments being approximately 13,000 years old. There are no areas of potential surface or subsurface subsidence, uplift, or collapse except for the low geologic deformation discussed in Section 1.4.3.6 of the SNF Project FSAR. With the exception of the loose, surficial, wind-deposited silt, soils are competent and form good foundations. Several geotechnical studies have been completed in and around existing tank farms. Liquefaction of soils beneath the tank farms is not a credible hazard because the water table is greater than 215 ft below ground surface. Liquefaction cannot occur in dry sediments.

A geotechnical investigation was performed at the site in 1989 to evaluate subsurface conditions for the design and construction of the HWVP. The results of that investigation are documented in Report 10805-385-016, *Report of Geotechnical Investigations for the Proposed Hanford Waste Vitrification Plant, Hanford Washington* (Dames and Moore 1989). The HWVP Project was canceled and the site is being used for the CSB. Subsurface conditions were investigated by using 17 borings ranging in depth from 20 to 100 ft at the site and in the surrounding area. Geophysical tests, including a series of downhole seismic tests, were conducted in boring VP-8, approximately 400 ft from the CSB. Borehole VP-15 (Figure A1-13) is at the CSB site.
The 17 geotechnical investigation borings were advanced without drilling fluid by driving a 6-in.-diameter, steel "core barrel" with a "split jar" downhole hammer. As the borings were advanced, a 6-in.-diameter steel casing was driven to prevent caving of soils above the sampling depth. Water was occasionally poured into the hole to prevent caving of the loose, dry soils and to aid in recovering samples. Slightly disturbed soil samples were taken at 5-ft intervals by driving a sampler using either a 705-lb hammer falling through a distance of 18 in. or a 825-lb hammer falling through a distance of 28 in. During the sampling process, resistance to penetration of the sampler, in blows per foot, was recorded.

The soil profile in the upper 100 ft consists essentially of the three strata shown in Figure A1-14. Borehole VP-14 is approximately 230 ft west of the CSB. The stratigraphy in VP-15 (Figure A1-13) is the same as at VP-14 (Figure A1-14). The dynamic soil properties of the site are summarized in Figure A1-15. More detailed descriptions of the geotechnical investigations and interpretations are found in Report 10805-385-016 (Dames and Moore 1989).

A1.5 NATURAL PHENOMENA THREATS

Table A1-1 (from WHC-SD-SNF-DB-009, Table 1) summarizes the natural phenomena design loads to be used for the design of safety-class structures, systems, and components (SSCs) for the CSB. The industry codes and standards used for the design, fabrication, and procurement of safety-significant SSCs are provided in Chapter A4.0. The design codes used for general service equipment follow industry-acceptable design codes and standards. These SSCs are designed to codes, standards, regulations, and orders as listed in Section 1.2 of the SNF Project FSAR. Each of these phenomena and the bases for the analysis values have been discussed in Section 1.4 of the SNF Project FSAR and Section A1.4.

A1.6 EXTERNAL HUMAN-GENERATED THREATS

This section identifies and investigates specific potential human-generated threats to CSB operation. Threats to the CSB from human activities that are not known at this time will be evaluated by the Unreviewed Safety Question process when they are identified.

A1.6.1 Aircraft Activity

See Section 1.6.1 and Figure 1-35 of the SNF Project FSAR for the methodology used in aircraft activity analysis. According to Assessment of Aircraft Impact on the Canister Storage Building and Cold Vacuum Drying Facility (Beary 1997), there are nine active airports within a 24-mi radius of the CSB. Eight of these are small airports that serve only general aviation aircraft. The Richland Airport, 21 mi southeast of the CSB, supports primarily general aviation operations, but two commercial freight carriers per day land and take off from the airport's
Table A1-1. Canister Storage Building Safety-Class Natural Phenomena Design Loads.

<table>
<thead>
<tr>
<th>Hazard</th>
<th>Load</th>
<th>Design guidance</th>
</tr>
</thead>
<tbody>
<tr>
<td>Seismic</td>
<td>Median response spectra:</td>
<td>DOE Order 5480.28&lt;sup&gt;8&lt;/sup&gt;</td>
</tr>
<tr>
<td></td>
<td>0.35 g horizontal</td>
<td></td>
</tr>
<tr>
<td></td>
<td>0.23 g vertical</td>
<td>DOE-STD-1020-94&lt;sup&gt;8&lt;/sup&gt;</td>
</tr>
<tr>
<td>Straight wind</td>
<td>80 mi/h, fastest mile at 30 ft</td>
<td>ASCE 7-93&lt;sup&gt;4&lt;/sup&gt;</td>
</tr>
<tr>
<td></td>
<td></td>
<td>DOE-STD-1020-94&lt;sup&gt;8&lt;/sup&gt; (including missiles)</td>
</tr>
<tr>
<td>Tornado</td>
<td>Wind speeds</td>
<td>NRC Standard Review Plan&lt;sup&gt;2&lt;/sup&gt;</td>
</tr>
<tr>
<td></td>
<td>200 mi/h total</td>
<td>3.3.2 Tornado Loading</td>
</tr>
<tr>
<td></td>
<td>160 mi/h rotational</td>
<td></td>
</tr>
<tr>
<td></td>
<td>40 mi/h translational</td>
<td></td>
</tr>
<tr>
<td>Volcanic ash</td>
<td>24 lb/ft&lt;sup&gt;2&lt;/sup&gt; ground ash load</td>
<td>NRC Standard Review Plan&lt;sup&gt;2&lt;/sup&gt;</td>
</tr>
<tr>
<td></td>
<td></td>
<td>3.8.4 Other Seismic Category Structures</td>
</tr>
<tr>
<td>Flooding</td>
<td>Dry site for river flooding</td>
<td>ANSI/ANS-2.8-1992&lt;sup&gt;4&lt;/sup&gt;</td>
</tr>
<tr>
<td></td>
<td>Site drainage basin: 7.4 in. for</td>
<td></td>
</tr>
<tr>
<td></td>
<td>6-hour probable maximum precipitation</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Site drainage: 9.2 in. for</td>
<td>NRC Standard Review Plan&lt;sup&gt;2&lt;/sup&gt;</td>
</tr>
<tr>
<td></td>
<td>6-hour probable maximum precipitation</td>
<td>2.4.2 Floods</td>
</tr>
<tr>
<td>Lightning</td>
<td>Lightning protection shall be</td>
<td>NFPA 780&lt;sup&gt;8&lt;/sup&gt;</td>
</tr>
<tr>
<td></td>
<td>provided for facility as required by</td>
<td></td>
</tr>
<tr>
<td></td>
<td>NFPA 780&lt;sup&gt;8&lt;/sup&gt;</td>
<td></td>
</tr>
<tr>
<td>Snow</td>
<td>20 lb/ft&lt;sup&gt;2&lt;/sup&gt; ground load</td>
<td>ASCE 7-93&lt;sup&gt;4&lt;/sup&gt;</td>
</tr>
</tbody>
</table>

<sup>2</sup>DOT Order 5480.28, Natural Phenomena Hazards Mitigation, U.S. Department of Energy, Washington, D.C.
<sup>7</sup>NFPA 780, 1995, Lightning Protection Systems, National Fire Protection Association, Quincy, Massachusetts.

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Annex A — Canister Storage Building

The nearest airport with significant commercial and military air activity is the Tri-Cities Airport, 29 mi southeast of the CSB. Table A1-2 lists the airports within 24 mi of the CSB and their distance from and orientation to the facility.

Table A1-2. Airports within Twenty-Four Miles of the Canister Storage Building.

<table>
<thead>
<tr>
<th>Airport</th>
<th>Distance from CSB (statute miles)</th>
<th>Direction from CSB</th>
<th>Type of operation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Green Acres</td>
<td>13</td>
<td>East</td>
<td>Private</td>
</tr>
<tr>
<td>Mattawa Air Strip</td>
<td>14</td>
<td>North-northwest</td>
<td>Private</td>
</tr>
<tr>
<td>Christenson Brothers</td>
<td>15</td>
<td>Northwest</td>
<td>Private</td>
</tr>
<tr>
<td>Wahluke Strip</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>McWhorter Ranch</td>
<td>16</td>
<td>South</td>
<td>Private</td>
</tr>
<tr>
<td>Dorman Field</td>
<td>18</td>
<td>East</td>
<td>Private</td>
</tr>
<tr>
<td>Basin City Airfield</td>
<td>19</td>
<td>East</td>
<td>Private</td>
</tr>
<tr>
<td>Desert Aire</td>
<td>19</td>
<td>West-northwest</td>
<td>Commercial</td>
</tr>
<tr>
<td>Richland Airport</td>
<td>21</td>
<td>Southeast</td>
<td>Commercial</td>
</tr>
<tr>
<td>Slinkard Airfield</td>
<td>23</td>
<td>East</td>
<td>Private</td>
</tr>
</tbody>
</table>

CSB = Canister Storage Building.

DOE-STD-3014-96, *Accident Analysis for Aircraft Crash into Hazardous Facilities*, provides the methodology to conservatively evaluate and assess the significance of aircraft crash risk on facility safety. The CSB meets the standard's guidelines for applicability because it contains enough radioactive material to be classified as a hazard category 2 nuclear facility according to the criteria established in DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*.

The approach used in DOE-STD-3014-96 includes the methodology to evaluate the frequency of aircraft impact into the facility. If the frequency of aircraft impact, calculated according to the methodology given in the standard, exceeds $10^6$/yr, the analysis proceeds to a consideration of whether those aircraft that have a high impact frequency could actually cause damage resulting in potential releases from the facility. Using that methodology, the frequency of aircraft impact into the CSB has been assessed (Beary 1997).

The methodology provided in DOE-STD-3014-96 considers two contributors to the overall frequency of aircraft crashes: (1) the frequency of crashes resulting from nearby airport activity (i.e., takeoffs and landings) and (2) the frequency of crashes during overflights of the facility.
Only runways within about 24 mi of the facility are included on the crash location probability tables given in Appendix B of DOE-STD-3014-96. Activities on runways at greater distances are not considered potential contributors to the frequency of a crash at the facility location for any category of aircraft. For calculating the frequency of crashes from aircraft flying over the site, DOE-STD-3014-96 provides DOE site-specific frequency data, in crashes per square mile per year centered at the site, for various classes of aircraft.

As stated above, there are 9 active airports within a 24-mi radius of the CSB (Beary 1997). All operations at these airports involve general aviation aircraft. The nearest airports serving commercial or military aircraft are the Desert Aire Airport, 19 mi west-northwest of the CSB, the Richland Airport, 21 mi southeast of the CSB, and the Tri-Cities Airport 29 mi southeast. Table A1-2 lists the 9 airports within 24 mi of the CSB and their distances from and orientation to the facility.

For nearby airports and the overflight activities, DOE-STD-3014-96 bases the calculation of frequency of aircraft crash into a facility on a "four factor formula" that considers the following:

1. The number of aircraft operations at nearby airports and overflying the site (N)
2. The probability that an aircraft will crash (P)
3. The conditional probability that, given a crash, the aircraft crashes into a one-square-mile area where the facility is located (f(x,y))
4. The site-specific effective area of the facility (Aₑ).

The formula is applied individually to each category of aircraft for both the nearby airports and the overflight activities. The overall frequency of aircraft crashes into the facility is the sum of the frequencies calculated for all categories of aircraft and for all operational modes.

The distance and orientation of the runways for the nine nearby airports were compared with tables in Appendix B of DOE-STD-3014-96 to identify the flight sources that contribute to the frequency of an aircraft crashing in a one-square-mile area in the vicinity of the CSB. The screening revealed that all of the runways are too far from the CSB to contribute to the crash frequency for nearby airport operations. Therefore, nearby airport activities do not contribute significantly to the overall frequency of an aircraft crash impacting the CSB.

Multiplying the NPf(x,y) factors by the effective impact areas gives the annual frequency of impact into the facility for each category of aircraft. These values are given in Table A1-3.
Table A1-3. Overflight Crash Frequencies for the Canister Storage Building.

<table>
<thead>
<tr>
<th>Category</th>
<th>Np(x,y) (Impacts/yr/mi²)</th>
<th>A_eff (mi²)</th>
<th>Crash frequency (per year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>General aviation</td>
<td>$1 \times 10^{-4}$</td>
<td>$9.42 \times 10^{-3}$</td>
<td>$9.42 \times 10^{-7}$</td>
</tr>
<tr>
<td>Commercial aviation, air carrier</td>
<td>$1 \times 10^{-7}$</td>
<td>$3.33 \times 10^{-2}$</td>
<td>$3.33 \times 10^{-9}$</td>
</tr>
<tr>
<td>Commercial aviation, air taxi</td>
<td>$1 \times 10^{-6}$</td>
<td>$3.11 \times 10^{-2}$</td>
<td>$3.11 \times 10^{-8}$</td>
</tr>
<tr>
<td>Military aviation, large aircraft</td>
<td>$1 \times 10^{-7}$</td>
<td>$3.00 \times 10^{-2}$</td>
<td>$3.00 \times 10^{-9}$</td>
</tr>
<tr>
<td>Military aviation, small, low performance</td>
<td>$4 \times 10^{-8}$</td>
<td>$1.53 \times 10^{-2}$</td>
<td>$6.12 \times 10^{-10}$</td>
</tr>
<tr>
<td>Total</td>
<td></td>
<td></td>
<td>$9.8 \times 10^{-7}$</td>
</tr>
</tbody>
</table>

The major contributor to the frequency of aircraft impact into the CSB is general aviation aircraft during in-flight operations. The overall frequency of aircraft impact from all sources is $9.8 \times 10^{-7}$/yr. DOE-STD-3014-96 specifies that if the total impact frequency, calculated according to the method it gives, is less than $10^{6}$/yr, the safety risk is below the level of concern. The calculated impact frequency for the CSB is very close to the evaluation guideline given in DOE-STD-3014-96. Therefore, for extra margin, a structural evaluation of safety-class structural features of the CSB was conducted.

General aviation was the only aircraft category that contributed significantly to the overall crash impact frequency. The selection of representative aircraft in the general aviation category was based on conversations with Federal Aviation Administration personnel at the Tri-Cities Airport. It was estimated that approximately 70% of the general aviation aircraft involved in airport operations at that airport were single-engine aircraft, with the remaining 30% dual-engine aircraft. An upper limit for size of general aviation aircraft was reported to be 12,500 lb.

Based on these data, a Cessna 172 single-engine aircraft and a dual-engine Beechcraft B200 were chosen to represent the general aviation category of aircraft for the structural analysis (Beary 1997). Using the methodology presented in Appendix C of DOE-STD-3014-96, an evaluation basis kinetic energy hazard was calculated to be $3.58 \times 10^{6}$ ft-lb. The evaluation determined that the energy generated by a Cessna 172 impact is less than the evaluation basis kinetic energy. Therefore, the analysis of safety structural features was carried out for only the impact of the Beechcraft 200 aircraft.
The safety-class structural features of the CSB are the operating deck, the air intake structure for vault ventilation, and the exhaust stack. The operating deck is a horizontal structure. The analysis conservatively assumes that the missile that impacts the operating deck falls from a direct vertical position. The vertical impact velocity of the Beechcraft 200, calculated from the evaluation basis kinetic energy and the weight of the aircraft, is 140.4 ft/s. The critical missile selected was the Beechcraft 200 engine.

The effective fly-in area of the stack was included as a separate entity in the summation of the total $A_{ef}$ for the CSB. The effective stack skid area is covered by an allowance in the building dimensions. The values of WS, cot $\phi$, and $S$ are given in tables B-16, B-17 and B-18 of DOE-STD-3014-96 for the categories of aircraft. For the military aviation categories, different values for the mean of cot $\phi$ and the mean skid distance are given for takeoff and landing. DOE-STD-3014-96 recommends using the takeoff values for the calculations involving nonairport operations. Table A1-4 presents the calculated effective areas for the CSB for all the categories of aircraft.

Table A1-4. Results of Effective Area Calculations for the Canister Storage Building.

<table>
<thead>
<tr>
<th></th>
<th>General aviation</th>
<th>Commercial aviation</th>
<th>Military aviation</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Air carrier</td>
<td>Air taxi</td>
</tr>
<tr>
<td>WS</td>
<td>50</td>
<td>98</td>
<td>59</td>
</tr>
<tr>
<td>R, ft (CSB)</td>
<td>292.54</td>
<td>292.54</td>
<td>292.54</td>
</tr>
<tr>
<td>R, ft (stack)</td>
<td>9.9</td>
<td>9.9</td>
<td>9.9</td>
</tr>
<tr>
<td>H, ft (CSB)</td>
<td>56</td>
<td>56</td>
<td>56</td>
</tr>
<tr>
<td>H, ft (stack)</td>
<td>75</td>
<td>75</td>
<td>75</td>
</tr>
<tr>
<td>cot $\phi$</td>
<td>8.2</td>
<td>10.2</td>
<td>10.2</td>
</tr>
<tr>
<td>L, ft (CSB)</td>
<td>259</td>
<td>259</td>
<td>259</td>
</tr>
<tr>
<td>L, ft (stack)</td>
<td>7</td>
<td>7</td>
<td>7</td>
</tr>
<tr>
<td>W, ft (CSB)</td>
<td>136</td>
<td>136</td>
<td>136</td>
</tr>
<tr>
<td>W, ft (stack)</td>
<td>7</td>
<td>7</td>
<td>7</td>
</tr>
<tr>
<td>S, ft</td>
<td>60</td>
<td>1440</td>
<td>1440</td>
</tr>
<tr>
<td>$A_t+A_s+A_{sk}$</td>
<td>0.00942</td>
<td>0.03328</td>
<td>0.03106</td>
</tr>
</tbody>
</table>

CSB = Canister Storage Building.
cot $\phi$ = mean of the cotangent of the aircraft impact angle.
$S$ = aircraft skid distance.
WS = aircraft wingspan.
For the air intake structure, an upright, reinforced concrete structure 2 ft thick, the analysis assumes that the missile path is horizontal before impact. The horizontal impact velocity, calculated from the evaluation basis kinetic energy and the weight of the aircraft, is 74.9 ft/s.

The analysis (Beary 1997) assumes that the velocity of the engine after it is detached from the plane is equal to the velocity calculated for the plane. The structural analysis shows that both the operating deck thickness, 5 ft of reinforced concrete, and the thickness of the air intake structure's vertical wall are more than adequate to withstand the impact of the evaluation basis aircraft missile.

See Section 1.6.1 of the SNF Project FSAR for an analysis of the risk to the CSB from helicopters.

A1.6.2 Other Transportation Accidents

See Section 1.6.2 of the SNF Project FSAR for a discussion regarding guidance for evaluating transportation accidents.

For the purposes of this evaluation, the CSB critical plant structures of concern, which are similar to those in a nuclear power plant that are required for safe plant operation and shutdown, are the below-grade structure (i.e., the base slab, interior and exterior vault walls, and the intake and exhaust plenums), operating deck, and the intake and exhaust foundation bases. These are all reinforced concrete structures that are designed to withstand the tornado pressure drop of 0.9 lb/in². It is reasonable to assume that they would also withstand a 1 lb/in² overpressure (see Section 1.6.2 of the SNF Project FSAR).

The CSB operating shelter is not considered a critical plant structure relative to the need to maintain the integrity of its shell as it is not credited for limiting offsite releases. However, it is also designed to withstand the tornado differential pressure of 0.9 lb/in² as failure of the shelter may adversely impact safety-class SSCs.

NRC Regulatory Guide 1.91, Evaluation of Explosions Postulated to Occur on Transportation Routes near Nuclear Power Plants, defines the safe distance, R, in meters, as

\[ R \geq 18 \times W^{0.6} \]

where

\[ W = \text{kg of TNT}. \]
From this correlation, the safe distances for the truck and railcar capacities listed in NRC Regulatory Guide 1.91 are as follows:

- 1,683 ft for a truck (50,600 lb capacity)
- 2,316 ft for a single railcar (60,000 kg capacity).

Highway 240 is 5 mi from the CSB. The nearest railroad not controlled by DOE is at the 1100 Area located approximately 20 mi south of the CSB. DOE currently owns and controls the railroad north of this point as discussed in Section 1.3.1 of the SNF Project FSAR. At these distances, explosive shipment on roads and railroads not controlled by DOE do not represent a threat to the CSB.

The main roadway and railroad controlled by DOE that pass nearest to the CSB are Route 4, which passes 1,155 ft to the west, and the mainline of the Hanford Railroad, which passes 9,000 ft to the north. The mainline of the Hanford Railroad is outside the above-listed safe distance, but Route 4 is not.

NRC Regulatory Guide 1.91 allows for a risk assessment when the safe distance criterion is not met. If the exposure rate is less than $10^7$ events/yr by best estimate analysis or less than $10^6$ events/yr by conservative analysis, then the risk of damage being caused by explosion is sufficiently low. The exposure rate, $r$, is defined as follows:

$$r = n \times f \times s$$

where

- $n$ = explosion rate for the substance and transportation mode in question, per kilometer
- $f$ = frequency of shipment for the substance in question in shipments per year
- $s$ = exposure distance in km. This is the length of the roadway that would be within the safe distance of any portion of the facility. It is a function of the building dimension parallel to the roadway, the setback of the building from the roadway, and the calculated safe distance, $R$.

H&R 522-1, Recommended Onsite Transportation Risk Management Methodology, concludes that a Hanford Site truck accident rate of $5.5 \times 10^{-8}$ accident/mi should be used for risk analysis. This study was performed for shipment of radioactive materials and took credit for risk reduction factors that would be associated with such shipments. These included driver participation in special safety programs, the road conditions north of the Hanford Site Wye Barricade, and making shipments during off-peak traffic hours. A risk reduction factor of 40 was obtained for Hanford Site two-lane roads such as Route 4 (H&R 522-1). This risk reduction factor also would be appropriate for shipment of large quantities of explosive materials. For this analysis, it is conservatively assumed that the conditional probability that an explosion will result...
as a consequence of an accident is 0.1. Therefore, the explosion rate, \( n \), is \( 5.5 \times 10^{-9} \) explosions/mi.

There are currently no routine shipments of explosives on Route 4 of a quantity that would place the CSB at risk. However, such shipments could occur during the facility's lifetime. For this analysis, the number of such shipments per year, \( f \), was conservatively assumed to be six. The exposure distance, \( s \), for the CSB was calculated to be 0.51 mi based upon a building length of 257 ft, a setback from Route 4 of 1,155 ft, and a safe distance of 1,683 ft.

With these data, the exposure rate, \( r \), was conservatively calculated to be \( 4.36 \times 10^{-6} \) events/yr. As this is less than \( 10^{-4} \) events/yr, no additional analysis is required.

Based upon the above analyses, it is concluded that explosive shipments on roadways and railways, controlled and not controlled by DOE, do not represent a risk to the CSB.

A1.7 NEARBY FACILITIES

Accidents in certain facilities in the 200 East Area (Figure A1-2) have the potential to impact the CSB facility and its operations, as discussed in Section A1.7.1. Conversely, certain nearby facilities can potentially be affected by accidents in the CSB, as discussed in Section A1.7.2.

Threats to the CSB from nearby facilities that are not known at this time will be evaluated using the Unreviewed Safety Question process when they are identified. This also applies to those activities that have been identified but that may change significantly in terms of a potential increased risk to the facility.

A1.7.1 Potential Effects from Nearby Facilities

Potential hazards to the CSB from onsite or offsite hazardous operations or facilities are examined under three general classifications:

- Nonreactor nuclear and nonnuclear industrial facilities within 5 mi of the CSB including all activities conducted in and near the 200 East Area
- Nuclear reactors within a 50-mi radius of the CSB
- Military activities.

Events at a nearby facility that challenge CSB habitability will be defined in the CSB's emergency response plan and emergency response procedures (i.e., notifications, take-cover, emergency
shutdown, excavation). See Section A15.4.6 and Section 15.4.6 of the SNF Project FSAR for additional information on protective actions.

A1.7.1.1 Hazards to the Canister Storage Building from Nonreactor Nuclear Facilities. Facilities currently operating, recently operating, or with potential to operate at the 200 East and 200 West Areas were screened along with the area between the 200 East and 200 West Areas. For this FSAR, selected facilities were those believed to pose the most risk to safe operations at the CSB. Safety analysis reports and accident analyses prepared for these facilities were reviewed to determine possible hazards, such as radiological doses to personnel resulting from direct radiation, release of airborne radioactivity, or exposure to toxic chemicals.

Considered, but not included in this FSAR, were the 200 East Area Burial Grounds, the Liquid Effluent Retention Facility, and the 200 Areas Effluent Treatment Facility in the 200 East Area. In the 200 West Area, the T Plant, U Plant, Reduction-Oxidation Plant, and the 222-S Laboratory were considered, but not included. These facilities have insufficient radiological or toxicological inventories in a dispersible form to represent a risk to the CSB operation.

The specific facilities discussed here include the Interim Storage Area (ISA), Plutonium-Uranium Extraction (PUREX) Facility, the Grout Treatment Facility, B Plant, the Waste Encapsulation and Storage Facility (WESF), the River Protection Project (RPP) facilities, 242-A Evaporator/Crystallizer, Plutonium Finishing Plant (PFP), and the Low-Level Waste Disposal Site. The worst-case scenarios for each of these facilities may challenge CSB habitability. Requirements for safely performing such actions are addressed in the CSB emergency response procedures. During much of the CSB's 40-year life, the facility will not be particularly sensitive to these hazards as it will be simply a storage facility and will require little human supervision.

200 Area Interim Storage Area. The 200 Area Interim Storage Area (200 Area ISA) provides for the interim storage of non-defense reactor SNF housed in aboveground dry cask storage systems. The 200 Area ISA is located immediately to the west of the CSB.

The 200 Area ISA is a passive storage facility designed to hold 60 storage containers of SNF for up to 40 years. Potential hazards associated with the unloading and storage of these containers are the release of radioactivity, loss of shielding, or loss of configuration control for criticality. Seven design basis accidents were identified and analyzed to evaluate the potential consequences of these hazards to onsite workers and the public. The seven design basis accidents analyzed are: (1) cask handling/drop, (2) mobile crane mechanical failure, (3) cask tip over, (4) fuel rod rupture, (5) seismic event, (6) tornado wind, and (7) fire.

In all but one case, the consequences for all three fuel types were prevented by the passive design features of the storage systems. In the case of the mobile crane mechanical failure, credit
was not taken for the integrity of the Neutron Radiography Facility TRIGA\textsuperscript{1} storage casks and an unmitigated release was assumed. The consequences for this accident were found to be significantly below both the offsite limits and the onsite guidelines. However, if CSB habitability is challenged by an event at the 200 Area ISA, the appropriate actions (i.e., notification, take-cover, emergency shutdown, evacuation) will be defined in the CSB's emergency response procedures.

**Plutonium-Uranium Extraction Facility.** The PUREX Facility is located in the 200 East Area, 1.5 mi east-southeast of the CSB. It is the most recently constructed of the irradiated fuel separation facilities and was used for processing N Reactor fuel. The principal product was a solution of plutonium nitrate that was transferred to the PFP for further processing. Another product was uranyl nitrate solution, which was processed at the Uranium Oxide Plant. The facility has been cleaned out to the extent practical and is in transition to surveillance and maintenance mode awaiting a final decision on disposal of the facility. The processing canyon area and the tunnels still contain airborne contamination from previous processing activities. If the status of PUREX changes (i.e., active decontamination and decommissioning), potential impacts to the CSB will be re-evaluated.

The bounding accident for the PUREX facility in the surveillance and maintenance mode is a seismic event that releases a portion of the residual contamination in the ventilation gallery above the canyon. This accident would not impair the ability of CSB personnel to perform required safety actions because the quantity of material available for release is small.

**Grout Treatment Facility.** The Grout Treatment Facility is located on the eastern perimeter of the 200 East Area approximately 2.7 mi east of the CSB. The Grout Treatment Facility combines tank wastes with grout-forming solids to form a grout slurry. The waste feed stream constituent of this slurry consists of low-level fractions of radioactive wastes. The slurry ultimately is pumped into near-surface, concrete-lined vaults for permanent disposal. Only one vault has been filled with grout (completed in 1987) even though additional vaults have subsequently been designed and constructed. This facility is currently not operational but has the potential to operate again in the future.

If the Grout Treatment Facility resumes operation, the maximum credible accident release postulated involves a double-ended jumper leak on the grout feed line with the pit cover blocks left off. This postulated release would spray low-level waste as an aerosol for a maximum period of 24 hours before detection by a visual inspection. If the status of the Grout Treatment Facility changes (i.e., it begins operations), potential impacts to the CSB will be re-evaluated.

**B Plant Facility.** The B Plant is located approximately 1,500 ft east of the CSB. Until 1952, B Plant was operated as a fuel separation facility. In 1968 it was converted to a waste fractionation plant to remove $^{137}$Cs and $^{90}$Sr from radioactive waste streams. This had the effect

\textsuperscript{1}TRIGA is a registered trademark of Gulf General Atomics Company, Incorporated.
of reducing heat loads in the double-shell tanks. The B Plant currently is deactivated and in
transition to surveillance and maintenance mode.

The worst-case credible accident at B Plant is a postulated flooding of the 291-B high-
efficiency particulate air filters and a subsequent hydrogen explosion. According to
HNF-SD-WM-BIO-003, *B Plant Basis for Interim Operation*, the unmitigated dose to the
collocated worker at 100 m from B Plant was calculated to be 952 rem and the maximum offsite
dose to be 0.368 rem at the Columbia River, 27 km east of B Plant. Systems and administrative
controls are in place at B Plant to mitigate the releases. However, this scenario has the potential
to require evacuation of CSB operating personnel. As stated in Section A1.7.1, if CSB
habitability is challenged by an event at a nearby facility, such as B Plant, the appropriate actions
(i.e., notifications, take-cover, emergency shutdown, evacuation) will be followed as defined in
the CSB’s emergency response procedures.

**Waste Encapsulation and Storage Facility.** The WESF is distinct from B Plant, even
though it is located on the west end of B Plant and shares a common wall with the plant.
Historically WESF was involved in converting the cesium and strontium removed from waste
streams at B Plant into cesium chloride and strontium fluoride salts. These materials then were
encapsulated in double-walled metal containers and stored in a water-filled cooling basin.
Strontium fluoride and cesium chloride capsules are still being stored in this fashion at WESF, but
no new capsules are being produced.

According to HNF-SD-WM-BIO-002, *Waste Encapsulation and Storage Facility Basis
for Interim Operation*, the worst-case credible accident at WESF postulated involves a loss of
cooling water in the storage pool with a subsequent rupture of capsules and aerosolization of the
capsule contents. The maximum dose to the public receptor at the near bank of the Columbia
River is calculated to be 9 rem. The unmitigated dose at 100 m is 1,700 rem from inhalation and
32 rem from radiation shine. Systems and administrative controls are in place at WESF to
mitigate the releases. However, this is an accident that has the potential to require evacuation of
CSB personnel. As stated in Section A1.7.1, if CSB habitability is challenged by an event at a
nearby facility, such as the WESF, the appropriate actions (i.e., notifications, take-cover,
emergency shutdown, evacuation) will be defined in the CSB’s emergency response procedures.

**River Protection Project.** The RPP facilities include 149 single-shell tanks and 28 double-
shell tanks for the storage of liquid radioactive mixed waste solutions from the Hanford Site’s
chemical processing plants, and associated support facilities that include holding tanks, transfer
lines, and valve pits. The RPP facilities are located in both the 200 East and 200 West Areas.

The RPP facilities nearest to the CSB are the B, BX, and BY Tank Farms, located
approximately 0.8 mi northeast of the CSB. Each of the farms has 12 single-shell tanks. The
B and BX single-shell tanks are rated at 500,000 gal each, and those in the BY Tank Farm are
rated at 750,000 gal each. In addition, there are four 55,000-gal tanks in the B Tank Farm, and a
double-contained receiver tank serves the BX and BY Tank Farms. The tanks in these tank farms
were constructed between 1943 and 1949.
According to HNF-SD-WM-BIO-001, *Tank Waste Remediation System Basis for Interim Operation*, the worst-case accident postulated for the RPP facilities is the inadvertent pumping of liquid waste through an open valve into a valve pit and consequent overflow of the pit, which results in an unmitigated dose of 330 rem at 100 m. The postulated cause of the accident is misrouting of waste during transfer operations. Waste transfers into Hanford Site single-shell tanks are no longer allowed. Most of the tanks in these three farms have been interim stabilized, meaning that as much of the liquid has been removed from the waste as practicable, leaving solids with a minimum liquid content. Administrative controls are in place at the tank farms for early detection and mitigation of a release caused by mistransfer during waste transfer operations. Also considered was a flammable gas deflagration in a tank that could result in an unmitigated dose of 650 rem at 100 m. However, because of the site distance (0.8 mi) and the flammability controls, this scenario would not significantly impact the CSB. Even though the cross-site transfer line passes near the CSB, a leak from it is not considered a credible event because of its safety-class, encased-pipe design (HNF-SD-WM-BIO-001, Addendum 2).

If there were a release from an RPP accident, the immediate action for CSB would be to respond to the take-cover sirens. Depending on the specific conditions, follow-up actions could include emergency shutdown and evacuation per emergency response procedures.

Eventual retrieval of the tank wastes for final disposal is planned for the future. Activities involving waste removal, and decontamination and decommissioning of the tanks, will be evaluated for their effect on the CSB safety analysis at that time.

**242-A Evaporator/Crystallizer.** The 242-A Evaporator/Crystallizer is located 1.5 mi east of the CSB. The 242-A Evaporator/Crystallizer uses evaporative concentration to reduce the volume of liquid wastes (low-heat-generating tank wastes). The concentrated slurry, reduced in volume, is transferred and stored in an underground waste storage tank. The process condensate is routed to the Liquid Effluent Retention Facility for storage and treatment at the Effluent Treatment Facility.

The worst-case accident scenario reported in HNF-SD-WM-SAR-023, _242-A Evaporator Safety Analysis Report_, involved a release from a spray leak in the pump room and a failure of the exhaust system's high-efficiency particulate air filters. Although this event is considered extremely unlikely, the event was analyzed for safety-class determinations at the 242-A Evaporator/Crystallizer. The release from a spray leak is comprised of a liquid component and an aerosol component. The analysis assumed that the liquid component would remain in the pump room; however, the aerosol component would be released to the environment via the K1 exhaust stack. The maximum unmitigated dose consequence to the onsite individual at 100 m from the evaporator would be 14 rem committed effective dose equivalent. Such a scenario might require action to prevent exposure of CSB operators to the release (i.e., take cover). As stated in Section A1.7.1, if CSB habitability is challenged by an event at a nearby facility, such as the 242-A Evaporator/Crystallizer, the appropriate actions (i.e., notifications, take-cover, emergency shutdown, evacuation) will be defined in the CSB's emergency response procedures.
Plutonium Finishing Plant. The PFP is located near the western boundary of the Hanford Site in the 200 West Area, 5.0 mi west of the CSB. It converts plutonium nitrate solution to plutonium metal and performs plutonium handling and storage operations. Contaminated liquid waste streams from the PFP are routed to the tank farms. This facility is in transition from its previous special nuclear material processing mode to preparation for decontamination and decommissioning.

During its previous mission of processing special nuclear materials, plutonium-bearing miscellaneous materials were produced and still exist in the facility. These include pure and mixed oxides, fluorides, oxalates, silicates, and organic-based sludge and residues. The current PFP missions are defined in WHC-SD-CP-SAR-021, Plutonium Finishing Plant Final Safety Analysis Report, as (1) receiving, sorting, storing, and shipping of special nuclear material, (2) stabilizing the reactive materials that remain in the plant, (3) providing laboratory support for other Hanford Site facilities, (4) handling radioactive and mixed waste, and (5) shutdown facility surveillance. As part of the materials stabilization activities, preparations are being made for startup of a calcining operation. Revisions to the safety analysis report to include this operation are in progress.

The bounding accident for the PFP is a seismic event with a horizontal ground acceleration of 0.2 g. The unmitigated consequences for this event are calculated to be 15.2 rem committed effective dose equivalent at the nearest occupied facility in the worst-case direction, 1,800 ft west-northwest of PFP, and 0.3 rem at the Site boundary, 7.8 mi west. Because of the distance separating the CSB from the PFP, the release would not adversely affect CSB operations and would not require evacuation of CSB personnel.

Low-Level Waste Disposal Site. The commercial low-level waste disposal site operated by U.S. Ecology, Inc., is the only non-DOE industrial facility within 5 mi of the CSB (Figure A1-1). The disposal site is on land leased from Washington State located 1 mi southwest of the CSB. The low-level waste is buried in lined containers. Monitoring of groundwater and vegetation is performed as required by the facility's NRC operating license and environmental impact statement. This facility is not likely to have a significant accidental airborne radioactive release that could adversely affect the CSB.

A1.7.1.2 Hazards to the Canister Storage Building from Nonnuclear Hanford Site Facilities. A number of nonnuclear industrial facilities operating in the 200 Areas pose the potential for accidental fires, explosions, or releases of toxic fumes. These include the Essential Materials Warehouse (Building 275-EA), oil and paint storage buildings, fabrication shops, gas cylinder storage buildings, the spare parts and electrical warehouse, B Plant storage buildings, maintenance facilities, gasoline service stations, and the powerhouse complexes in each area (284-E and 284-W). Considering its location, Building 275-EA may have the potential to pose a risk to CSB operations and personnel and is detailed below. The 2,000-lb chlorine bottles located at Building 283-E have been removed and are no longer considered a hazard to the CSB.
Building 275-EA. Building 275-EA, the Essential Materials Warehouse, was constructed in 1955 and is located 1.5 mi east-southeast of the CSB. It is classified as an unprotected wood frame structure and is susceptible to collapse as a result of an external event (e.g., earthquake, wind, snow, or ash loading) or an internal event (e.g., forklift collision with a bearing wall, or fire). Building 275-EA currently stores more than 100 different types of potentially hazardous solids and liquids including acids, bases, solvents, fluorides, pesticides, and herbicides. Radioactive materials are not stored in this building.

The worst-case chemical release postulated in the safety analysis occurs following a building collapse whereupon 2,450 kg of 1,1,1-trichloroethane evaporates under adverse atmospheric conditions. The maximum concentration at the PUREX Facility gatehouse and environs is calculated to be 187 ppm, which is below the time-weighted average threshold limit value of 350 ppm (i.e., the concentration for this chemical that an industrial worker may be repeatedly exposed to without adverse effects) as given in Threshold Limit Values and Biological Exposure Indices for 1989-1990 (ACGIH 1989). Concentration levels at the CSB would be significantly less than this because of the 1.5-mi separation.

A1.7.1.3 Hazards to the Canister Storage Building from Nuclear Reactors. Three recently shut down reactors, the N Reactor, Fast Flux Test Facility (FFTF), and the Critical Mass Laboratories, no longer pose a threat to the CSB. The N Reactor is undergoing decontamination and decommissioning, the FFTF is in standby mode and may operate again in the future, and the Critical Mass Laboratories in the 200 East Area north of the PUREX Facility are currently being used as tank farm office areas.

UNI-M-90, N-Reactor Updated Safety Analysis Report, describes the N Reactor as a 4,000 MW, dual-purpose, pressure tube, light-water cooled, graphite-moderated reactor. It is located in the 100 N Area and is about 5.5 mi from the nearest northwest Hanford Site boundary. The N Reactor began operating in 1964 producing plutonium for the defense program and steam for electrical power generation. It was shut down in 1987 for safety improvements and then subsequently defueled and placed in cold standby in 1988. It is currently being decontaminated and decommissioned.

It is planned to remove the reactor building structures for the production reactors down to the reactor block and then cocoon the reactor block for 75-year safe storage.

The FFTF is a 400 MW, sodium-cooled, mixed-oxide-fueled, reactor that is currently in standby status with the fuel removed from the core. Sodium is kept circulating in the loops at 400 °F until a decision is made to either restart the reactor for future missions or decommission it. The FFTF is located in the 400 Area and is approximately 4.5 mi from the nearest Hanford Site boundary, which is to the east of the facility (Figure A1-1). Spent FFTF fuel packaged in double-walled casks with an inert gas cover is stored in the Interim Storage Area at the 400 Area.

According to HEDL-TI-7500-FSAR, Fast Flux Test Facility (FFTF) Final Safety Analysis Report, the bounding accidents for the FFTF in its current status involve (1) a liquid sodium spill
and (2) damage to a cask and its contents in the Interim Storage Area. The sodium spill scenario postulates a spill of 180,000 lb of activated liquid sodium. The 2-hour dose at 1.5 mi from the FFTF is 0.015 mrem. The dose at 4.5 mi from the facility for a 30-day exposure is 0.26 mrem. The maximum credible cask release at the Interim Storage Area postulated cracking in 100% of the fuel pins of one cask, with crushing and exposure of 1% of the fuel material. The dose at 100 m from the facility was 4.5 rem, the maximum dose at the Site boundary was 4.0 mrem. Because of distance from the CSB, these accidents do not pose a hazard to CSB operations.

The only operating nuclear reactor on the Hanford Site is Washington Nuclear Plant (WNP)-2 operated by Energy Northwest. The location of this reactor is shown in Figure A1-1. WNP-2 is an operating commercial nuclear power plant using a boiling-water reactor steam supply system. According to Docket No. 50-397, Final Safety Analysis Report, Washington Nuclear Power Plant No. 2, the design power level was increased to 3,486 MW in 1995. The reactor was designed by the General Electric Company and is designated as a BWR/5 with a Mark II containment.

By the requirements of Title 10, Code of Federal Regulations, Part 100, "Reactor Site Criteria" (10 CFR 100), the following are the maximum allowable doses for WNP-2:

<table>
<thead>
<tr>
<th>Location</th>
<th>Duration</th>
<th>Whole body dose</th>
<th>Thyroid</th>
</tr>
</thead>
<tbody>
<tr>
<td>Exclusion area boundary</td>
<td>2 hours</td>
<td>250 mSv (25 rem)</td>
<td>3,000 mSv (300 rem)</td>
</tr>
<tr>
<td>Low population zone</td>
<td>30 days</td>
<td>250 mSv (25 rem)</td>
<td>3,000 mSv (300 rem)</td>
</tr>
</tbody>
</table>

The exclusion area boundary for WNP-2 is 1,950 m and the low population zone distance is 3 mi. The CSB is located approximately 11 mi from WNP-2. Using the atmospheric diffusion guidance provided in NRC Regulatory Guide 1.3, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors, to estimate the dose reduction as a function of distance, it was determined that the 2-hour and 30-day doses at the CSB would be reduced by a factor of 20. Review of the site-specific meteorology provided in the WNP-2 Final Safety Analysis Report (Docket No. 50-397) shows there would be no significant reduction for the change in wind direction. The factor of 20 reduction for distance would result in a whole body dose of 12.5 mSv (1.25 rem) and a thyroid dose of 150 mSv (15 rem).

The expected dose that would be received at the CSB should a loss of coolant accident occur would be significantly less than this. NRC Regulatory Guide 1.3 requires an assumption that 25% of the radioactive iodine and all of the noble gases are released to the containment. In fact, the emergency core cooling system would prevent most of these releases as little fuel damage would occur as a result of the loss of coolant accident.
Energy Northwest plans to add an independent spent fuel storage installation on their leased property. The independent spent fuel storage installation would be licensed following the requirements in 10 CFR 72. According to 10 CFR 72.106, an individual located at the installation's controlled area boundary shall not receive a dose greater than 5 rem. At the CSB, this would result in a dose not exceeding 0.25 rem.

A1.7.1.4 Hazards to the Canister Storage Building from Industrial Facilities off the Hanford Site. There are no oil or gas pipelines in the vicinity of the CSB. The nearest major natural gas pipeline to the CSB site is about 29 mi away. A 20-in. gas transmission line of the Northwest Pipeline Corporation is located east and essentially parallel to U.S. Highway 395 between Pasco and Ritzville, Washington. According to Memo 042AGH.96, Contact with Local Organizations to Support CSB SAR Chapter I (Hosler 1996), a second pipeline system consisting of parallel 36-in. and 42-in. lines, owned by Pacific Gas Transmission Company, passes through Wallula, approximately 33 mi from the site. These distances eliminate any potential hazard to plant operations from a natural gas fire or explosion.

The nearest petroleum product storage tanks are located 38 mi from the site. These are 23-million-gallon capacity tanks at the Chevron Pipeline Company, and 21-million-gallon capacity tanks at the Tidewater Barge Lines. There are no plans to use a third petroleum storage facility at the Port of Pasco (Hosler 1996).

Located within the Richland city limits is the Siemens Power Corporation's Richland Engineering and Manufacturing Facility. All operational steps for the manufacture of nuclear fuel for light water reactors are conducted within the facility including the conversion of UF₆ to UO₂. According to Siemens Power Corporation Richland Engineering and Manufacturing Facility Emergency Plan (Siemens 1994), the most limiting postulated accident at this facility is a fire in the UF₆ cylinder storage area. Fusible plugs in twelve cylinders are assumed to melt causing the release of UF₆ that reacts with moisture in the air to form UO₂F₂ (solids in the form of uranyl fluoride hydrates) and 4HF (as hydrogen fluoride gas). UF₆ is a radiological hazard by inhalation. However, UF₆ is also a concern because of its chemical toxicity and the associated HF that can cause skin and eye burns and lung impairment. This accident results in a dose that exceeds 10 mSv (1 rem) out to 1.2 mi. The accident also exceeds the Emergency Response Planning Guideline-2 (ERPG) toxicology limits out to 8.8 mi (ACGIH 1989). The ERPG-2 is the maximum airborne concentration below which it is believed that 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 their abilities to take protective action. The CSB is located 19 mi from the Siemens facility. Therefore, CSB operators would not be placed at risk by an accident at the Siemens facility.

No other nonnuclear industrial facilities or operations have been identified that may impact CSB operations.

A1.7.1.5 Hazards to the Canister Storage Building from Military Facilities. The Yakima Training Center is a subinstallation under the command of Fort Lewis (Tacoma, Washington).
Further information is given in the Final Environmental Impact Statement — Ft. Lewis Military Installation (DOA 1979). The southeastern boundary of the Yakima Training Center is located about 23 km (14 mi) from the CSB (see Figure A1-16). The Yakima Training Center is used for military maneuvers and weapons training and is the only significant military activity in the vicinity of the Hanford Site.

The only weapon currently in use at the Yakima Training Center known to present a hazard to the Hanford Site is the Multiple Launch Rocket System (MLRS). With a range of approximately 16 mi, the MLRS could potentially impact the CSB site. However, the MLRS is only fired from the perimeter of the Yakima Training Center into a centrally located impact zone. The safety fan for the MLRS is shown in Figure A1-16. The MLRS is fired away from the Hanford Site and is only fired with dummy warheads. Given this information, additional safety features, and the administrative controls in place at the Yakima Training Center, a weapons accident having an impact on the Hanford Site is very improbable.

A more probable hazard to Hanford Site facilities is a scenario in which a fire started within the Yakima Training Center boundary spreads to the Hanford Site. Exploding artillery shells, sparks from tracked vehicles or other machines, and careless smoking by troops might start brush fires that, under adverse meteorological conditions, could spread rapidly beyond the Yakima Training Center boundaries. The hazards associated with range fires were considered with other hazards in the hazard analysis and in Chapter A3.0 and were found to not impact the CSB adversely.

A1.7.2 Potential Effects to Nearby Facilities

Potential CSB accidents are discussed in Section A3.4.2. CSB hazard assessments, as discussed in Section A15.4.2, will use the analyses of these potential accidents to help develop the CSB emergency response plan and procedures, as discussed in Section 15.4 of the SNF Project FSAR and Section A15.4.

The hazard assessments for emergency planning for the CSB will characterize the potential consequences to workers, the public, and the environment for each postulated accident. The hazard assessments will also determine the emergency planning zone as well as the emergency class, protective actions, and the observable indications and criteria (emergency action levels) corresponding to the range of potential accidents.

The hazard assessments will also include information to determine potential impacts to nearby facilities. This information will be made available to nearby facilities for determination of appropriate actions to be included in their emergency response plans and procedures. Information contained in CSB emergency response plans and procedures will also include notifications, evacuation potential, operations impacts, radiation threats to workers, and other information relevant to nearby facilities. Prompt and accurate emergency notifications will be performed in accordance with DOE O 151.1, Comprehensive Emergency Management System.
DOE O 232.1A, *Occurrence Reporting and Processing of Operations Information*; applicable federal, state, or local requirements; and special agreements with offsite agencies or tribal governments, to mitigate the consequences and to protect the health and safety of workers, the public, and the environment.

**A1.8 VALIDITY OF EXISTING ENVIRONMENTAL ANALYSES**

See Section 1.8 of the SNF Project FSAR.

**A1.9 REFERENCES**


DOE Order 5480.28, Natural Phenomena Hazards Mitigation, U.S. Department of Energy, Washington, D.C.


Annex A — Canister Storage Building

Hosler, A. G., 1996, *Contact with Local Organizations to Support CSB SAR Chapter 1*


Annex A — Canister Storage Building


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Figure A1-1. Hanford Site Boundaries.
Figure A1-2. Onsite Population Distribution in the 200 East Area by Zone.
Figure A1-3. Wind Rose for the 200 East Area.

HMS Data from 1983 to 1991
(Rings are in 2% increments)

Figure A1-4. Wind Speed Histogram for the 200 East Area.
Figure A1-5. Hanford Site Topographic Map and Cross Section.
Figure A1-6. Geology of the 200 East and West Areas.
Figure A1-7. Stratigraphy of the 200 East Area.
Figure A1-8. Seismic Hazard Curves for the Canister Storage Building Location.
Figure A1-9. Contribution of the Various Folds to the Mean Hazard from the Yakima Fold.

200 East Area

- Yakima folds
- Umtanum-Gable Mountain
- Rattlesnake-Wallula Alignment
- Saddle Mountains
- Frenchman Hills
- Rattlesnake Hills
- Yakima Ridge
- Toppenish Ridge
- Manastash Ridge
- Hog Ranch
- Horse Heaven Hills

Mean annual frequency

Peak acceleration (g)
Figure A1-10. Contributions of the Crustal and Cascadia to the Mean Hazard at the 200 East Area.
Figure A1-11. Contributions of the Three Crustal Sources to the Mean Hazard from Crustal Sources at the 200 East Area.
Figure A1-12. Canister Storage Building Response Spectra Compared to the Performance Category 3 Response Spectra.
Figure A1-13. Log of Borings: Boring VP-15.

Key to Log Borings

- 32 Blows required to sampler one foot
- Indicates depth at which undisturbed sample was extracted
- Indicates depth at which disturbed sample was extracted
- Indicates sampling attempt with no recovery
Figure A1-14. Cross Section B - B': Subsurface Profile.
**Figure A1-15. Dynamic Soil Properties of the Canister Storage Building Site.**

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<th>Depth (ft)</th>
<th>Natural Density (lb/ft³)</th>
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<th>Shear Wave Velocity (ft/s)</th>
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* Estimated using data from previous investigations.

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CSB-1
Figure A1-16. The Location of Yakima Training Center with Respect to the Hanford Site.
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CHAPTER A2.0

FACILITY DESCRIPTION
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<thead>
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<tr>
<td>ALARA</td>
<td>as low as reasonably achievable</td>
</tr>
<tr>
<td>CAEM</td>
<td>continuous airborne effluent monitor</td>
</tr>
<tr>
<td>CAM</td>
<td>continuous air monitor</td>
</tr>
<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
</tr>
<tr>
<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
</tr>
<tr>
<td>DCS</td>
<td>distributed control system</td>
</tr>
<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
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<tr>
<td>FFTF</td>
<td>Fast Flux Test Facility</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>HEPA</td>
<td>high-efficiency particulate air (filter)</td>
</tr>
<tr>
<td>HPT</td>
<td>health physics technician</td>
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<tr>
<td>HVAC</td>
<td>heating, ventilation, and air conditioning</td>
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<tr>
<td>HWVP</td>
<td>Hanford Waste Vitrification Plant</td>
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<tr>
<td>IFMSF</td>
<td>Irradiated Fissile Material Storage Facility</td>
</tr>
<tr>
<td>MCC</td>
<td>motor control center</td>
</tr>
<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
</tr>
<tr>
<td>MHM</td>
<td>multi-canister overpack handling machine</td>
</tr>
<tr>
<td>NA</td>
<td>not applicable</td>
</tr>
<tr>
<td>NPH</td>
<td>natural phenomena hazard</td>
</tr>
<tr>
<td>NRC</td>
<td>U.S. Nuclear Regulatory Commission</td>
</tr>
<tr>
<td>PMP</td>
<td>probable maximum precipitation</td>
</tr>
<tr>
<td>RALS</td>
<td>roller arm limit switch</td>
</tr>
<tr>
<td>RH</td>
<td>relative humidity</td>
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<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
</tr>
<tr>
<td>SPSS</td>
<td>solenoid-powered switch striker</td>
</tr>
<tr>
<td>SSC</td>
<td>structure, system, and component</td>
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<tr>
<td>UPS</td>
<td>uninterruptible power supply</td>
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A2.0 FACILITY DESCRIPTION

A2.1 INTRODUCTION

This chapter provides facility and operating descriptions for the Canister Storage Building (CSB) (Building 212H) to support assumptions used in the hazard and accident analyses. These descriptions focus on all major facility features necessary to understand the hazard analysis and accident analysis, not just safety structures, systems, and components (SSCs). The level of detail used in this chapter is based on the significance of the preventive and mitigative features identified and the degree of complexity. The descriptions provide an understanding of the facility and processes in sufficient detail that a general understanding of the facility can be achieved without extensive consultation of the references cited. Chapter 2.0 of the Spent Nuclear Fuel (SNF) Project Final Safety Analysis Report (FSAR) provides an overview description of the SNF Project facilities and processes. This chapter provides a description of the CSB systems and processes but does not include information at the level of functional requirements and performance criteria because this information is provided in Chapter A4.0 for the CSB safety SSCs. The CSB systems and processes may be changed or modified as needed using the Unreviewed Safety Question process while protecting the key functions described in Chapter A4.0. Chapter A2.0 description changes of the systems and processes will be included in annual updates to the FSAR if not required to be submitted prior to implementation. This chapter reflects CSB design as of the date of this FSAR.

Chapter A3.0 describes the hazards, accident analyses, and safety classifications of the SSCs for the CSB, and Chapter A4.0 describes safety-class and safety-significant SSCs in greater detail and also identifies safety controls (the Technical Safety Requirements).

A2.2 REQUIREMENTS

The requirements that establish the basis for design are identified in Section 2.2 of the SNF Project FSAR. The following list describes how these requirements are applied to the CSB and includes additional requirements that apply to the CSB.

- Title 10, Code of Federal Regulations, Part 835, “Occupational Radiation Protection” (10 CFR 835). The shielding requirements for the CSB shall meet the ALARA (as low as reasonably achievable) guidelines established in this document. The overall source term is derived from the radionuclide composition of the SNF contained in a nominal multi-canister overpack (MCO); however, local shielding is provided for a single maximum MCO as described in HNF-S-0425, Performance Specification for the Spent Nuclear Fuel Canister Storage Building.

• DOE Order 5400.1, *General Environmental Protection Program*. This order establishes the U.S. Department of Energy (DOE) environmental protection program.

• DOE Order 5480.7A, *Fire Protection*. This order provides design criteria for fire protection systems for new DOE facilities. In particular, Section 9.b(3) of this order requires automatic fire suppression systems in all new structures with more than 5,000 ft² of floor area or with a maximum possible fire loss greater than $1.0 million. Because of the detrimental effects of inadvertent sprinkler activation on the SNF in the MCOs and the low quantities of combustible materials present, the contractor has requested and received an exemption from this order for the operating area over the vault, the load-in/load-out area, and the sampling/weld area. The exemption is documented in Letter 96-SFD-320, *Contract No. DE-AC06-96RL13200 — Project W-379, Spent Nuclear Fuel Canister Storage Building (CSB) Review and Approval of Exemption and Deviation Requests from Automatic Fire Sprinkler/Fire Suppression System Requirements* (Sellers 1996).

• DOE Order 5480.28, *Natural Phenomena Hazards Mitigation*. This order establishes mitigation requirements for natural phenomena hazards (NPHs) and target probabilistic performance goals based on the facility performance category. These requirements are identified in WHC-SD-SNF-DB-009, *Canister Storage Building Natural Phenomena Hazards*. Additional discussion of the NPH performance criteria is provided in Section 1.5 of the SNF Project FSAR.

• DOE Order 6430.1A, *General Design Criteria*. The main reference standards and guides for facility design are presented in Division 13, “Special Facilities.” DOE Order 6430.1A, Section 1320, “Irradiated Fissile Material Storage Facilities,” contains additional requirements for irradiated fissile material storage facilities (IFMSF). In particular, paragraph 1320-4, “Special Design Features,” requires a designed safety-class passive cooling air system that ensures an acceptable temperature for the stored material. HNF-4742, *Canister Storage Building Compliance Assessment DOE Order 6430.1A, General Design Criteria*, documents compliance to DOE Order 6430.1A. All future changes to the facility will be screened against the matrix using the Unreviewed Safety Question process to determine whether the proposed change affects any design aspect relied upon to show compliance to the order.

In Letter 95-SFD-167, *Implementation of K Basins Spent Nuclear Fuel Project (SNFP) Regulatory Policy* (Sellers 1995), DOE established the requirement for new SNF Project facilities to achieve “nuclear safety equivalency” to comparable U.S. Nuclear Regulatory Commission
(NRC)-licensed facilities. The SNF Project identified the NRC requirements that were to be met
during design and construction only in addition to existing applicable DOE requirements to
establish nuclear safety equivalency. These NRC requirements and the process used to identify
them are documented in HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward,
Additional NRC Requirements*, and WHC-SD-SNF-DB-009.

Title 10, *Code of Federal Regulations*, Part 72, “Licensing Requirements for the
Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste” (10 CFR 72),
Subpart F, “General Design Criteria,” and Subpart H, “Physical Protection,” provide general
design criteria for a SNF storage facility. These criteria are identified in HNF-SD-SNF-DB-003
and are incorporated into the design criteria.

HNF-SD-SNF-DB-003 and WHC-SD-SNF-DB-009 established the design requirements to
be met for the CSB to achieve NRC equivalency based on 10 CFR 72. This rule is used for the
licensing of independent SNF storage installations. Section 72.3, “Definitions,” defines SSCs that
are considered “important to safety.” Section 72.122, “Overall Requirements,” requires that the
design bases for SSCs important to safety reflect appropriate combinations of the effects of
normal and accident conditions and the effects of natural phenomena.

Additional requirements are identified in the following documents:

- WHC-SD-SNF-DB-004, *Spent Nuclear Fuel Project Seismic Design Criteria, NRC
  Equivalency Evaluation Report*


- ANSI/ASME N509, *Nuclear Power Plant Air-Cleaning Units and Components*

- ANSI/ASME N510, *Testing of Nuclear Air Cleaning Systems*

  Storage Facilities at Nuclear Power Plants*

- NRC Regulatory Guide 8.8, *Information Relevant to Ensuring that Occupational
  Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably
  Achievable*

  Design Loads for Facilities*. Note: These requirements were applied only to the
  base sections constructed for the Hanford Waste Vitrification Plant (HWVP) facility.
  This document has been replaced by guidance in HNF-PRO-097, *Engineering
  Design and Evaluation*. 
For the CSB, important-to-safety SSCs have been identified in accordance with 10 CFR 72.3. Once SSCs have been identified as having a function meeting the definition of important to safety, the requirements for SSCs important to safety specified in 10 CFR 72 are imposed. The following graded approach is applied to SSCs important to safety using the guidance provided in NUREG/CR-6407, *Classification of Transportation Packaging and Dry Spent Fuel Storage Systems*, as modified by HNF-SD-SNF-DB-003.

- **A — Critical to Safe Operation**
  
  SSCs in this category include those whose failure or malfunction could directly result in a condition adverse to public health and safety. Important-to-safety SSCs in this category have been classified as safety class using the definition in DOE Order 6430.1A, with the additional requirements therein.

- **Category B — Major Impact on Safety**
  
  SSCs in this category include those whose failure or malfunction could result in a condition adversely affecting collocated worker health and safety. Note that from the definition of Category C, Category B is understood to include events that could significantly damage the MCO without severe impact to public health and safety. SSCs in this category are classified as safety significant.

- **Category C — Minor Impact on Safety**
  
  SSCs whose failure or malfunction would not significantly reduce containment and would not be likely to create a situation adversely affecting public or collocated workers' health and safety. SSCs in this category are unclassified and termed general service.

The SSCs identified as important to safety in accordance with 10 CFR 72.3 are listed in Tables A4-1 and A4-9. These lists also contain the important-to-safety classification for each SSC as determined by Chapter A3.0. The important-to-safety classification is directly incorporated into the tables of preventive and mitigative safety function features identified at the end of each design basis accident scenario in Section A3.4.2. External human-generated threats are discussed in Section A3.3.2.3. Three hoists, the tent gantry hoist and the two sampling/weld station hoists, are the only general service equipment items that are classified important-to-safety Category C because the hoists can potentially drop equipment that could cause minor damage to the top of an MCO.

### A2.3 FACILITY OVERVIEW

This section includes a brief overview of the CSB facility configuration, and the basic processes performed therein. The CSB provides a facility for receiving, sampling, monitoring,
welding, staging, and interim storage of MCOs. The CSB includes the partially constructed below-grade structures of the canceled HWVP in its design. According to HNF-SD-SNF-SP-012, Additional Guidance for Including Nuclear Safety Equivalency in the Canister Storage Building and Cold Vacuum Drying Facility Final Safety Analysis Report, the CSB design should consider evolving Hanford Site uses, public health and safety issues, and probable future uses of Hanford Site lands to assist in meeting NRC equivalency requirement Item 24 in HNF-SD-SNF-DB-003.

A2.3.1 Facility Description

The SNF CSB comprises a steel-frame building that encloses the operating area, the load-in/load-out area, the sampling/weld area, and three, equally sized, below-grade concrete vaults covered by a concrete operating area deck (Figure A2-1). Support functions and equipment are housed in a steel-frame support area on the north side of the operations building. Only the northernmost vault (vault 1) is equipped with steel tubes for staging of mechanically sealed MCOs and storage of the MCOs having welded caps (normally two in each storage tube). A mechanically sealed MCO can be safely staged for at least 40 years (see Chapter A3.0).

An MCO received at the CSB contains five or six baskets of SNF from the 100 K Reactor SNF storage basins stacked one on top of another (Figure A2-2). Figure A2-2 depicts an MCO containing Mark IV fuel baskets. The Mark IA fuel baskets are shorter, and an MCO is designed to contain six Mark IA fuel baskets. As discussed in HNF-SD-SNF-DR-003, Multi-Canister Overpack Design Report, the stainless steel outer shell and bottom of the MCO and the mechanically sealed top shield plug have been designed to meet the Boiler and Pressure Vessel Code (ASME 1995) and to provide the primary long-term containment and confinement for the enclosed SNF. The shield plug shields workers against radiological hazards from the SNF inside of the MCO. The shield plug also provides access to the interior of the MCO using four access ports on the top of the shield plug assembly. The access ports provide the means for sampling those selected MCOs to be monitored at the CSB. The four access ports will be sealed at the Cold Vacuum Drying Facility (CVDF) using bolted cover plates before the MCO is transported to the CSB. The shield plug features an integral axisymmetric lifting ring. The MCO handling machine (MHM) grapple uses the lifting ring to raise the MCO out of the transportation cask, transport the MCO to a storage tube location or MCO sampling/weld station, and lower the MCO into the storage tube or sampling/weld station.

The MCO storage vault (vault 1) contains 220 standard storage tubes, each capable of staging or storing two MCOs stacked one on top of the other. Vault 1 also contains six additional overpack storage tubes on the west side, each capable of accommodating one MCO. Vault 1 is designed to contain SNF currently being stored in the K East and K West Basins. According to HNF-SD-SNF-SARR-005, Multi-Canister Overpack Topical Report, the basins contain 2,271 metric tons (total mass) of SNF. The average MCO is projected to contain 5.96 metric tons of SNF (based on 70% Mark IV and 30% Mark IA fuel [HNF-SD-SNF-SARR-005]). Each MCO will contain either Mark IV or Mark IA fuel, consistent
with the criticality analysis. Based on these values, all of the K Basins SNF will be contained in
approximately 400 MCOs. Vault 1, at capacity, could store up to 440 MCOs plus 6 MCOs in
overpack storage tubes. Operational uses for vaults 2 and 3 are not authorized by this FSAR.

The storage tubes are supported from the foundation base slab of the vault and are
accessed through embedments in the operating deck, which are normally closed with removable
tube plugs. The standard tube plugs provide shielding, a filtered vent, and connections for
sampling of the storage tube atmosphere. Overpack storage tube plugs contain (1) connections
for sampling, purging, and relieving of tube atmosphere; (2) a pressure gauge for surveillance of
tube pressure; and (3) a lock-down device for the overpack tube plug.

The MCO storage vault (vault 1) is cooled by natural convection through the above-grade,
dedicated concrete and steel inlet structure and exhaust air stack and the below-grade reinforced
concrete plenums. The CSB floor plan and interior elevations are shown in Figure A2-3.

Each standard storage tube contains both a bottom impact absorber and an intermediate
impact absorber to mitigate the consequences of a concentric MCO drop. Only one MCO and a
bottom impact absorber are to be stored in an overpack storage tube. Intermediate impact
absorbers are not required for the overpack storage tubes. Bottom impact absorbers mitigate
damage to the storage tube and limit damage to the MCO in the event of a drop, as discussed in
Section A3.4.2.1. When placing a second MCO in a standard storage tube, the intermediate
impact absorbers protect the MCO cap lifting ring on the bottom MCO and protect the top MCO
from internal damage in the event of a drop.

The sampling/weld area, on the south end of the operating area, contains equipment for
pressure checking and sampling the MCOs, and welding equipment (future) for installing a welded
cap on the MCOs.

A2.3.2 Site Grading and Storm Drains

The CSB storm drain and flood control components have been designed to accommodate
the probable maximum precipitation (PMP) event, in accordance with WHC-SD-SNF-DB-009
requirements and Chapter A1.0. Culverts under roadways allow passage of runoff from storms of
less than a 25-year return frequency. The site civil grading and drainage plan is shown on
Figure A2-4. The CSB has been designed for a specific design basis storm event, a 25-year return
frequency storm, and takes into account storms larger than the one used for design by evaluating
the path that overflows (larger-than-design flows) will take when the system capacity is exceeded.

The CSB criterion for the overflow path assessment is to ensure that the PMP event, as
listed in Table A1-1, does not allow water flow into the CSB. WHC-SD-SNF-DB-009 specifies
that 9.2 in. over a 6-hour period be used for the CSB design. This PMP is estimated to be greater
than a “1 million-year average return period storm.”
The design storm used for culverts, drainage ditches, and general storm drainage is the 25-year, 6-hour storm. Letter FRP-112, *SNF Canister Storage Building* (Bedell 1996), specifies that “Culverts will be sized to flow with an unsubmerged inlet during the 25 year return 6 hour duration storm.” HNF-S-0425 states that the prevailing environmental condition for rain is “1.52 cm/hr (0.60 in/hr) maximum.” This corresponds to a 25-year storm, 1-hour duration.

The site and its surrounding tributary areas were analyzed and evaluated in CSB-C-0001, *SNF CSB Storm Drainage Calculation*, using the PMP criteria to determine the impact on the safety of the facility and facility workers. No non-CSB area contributed PMP drainage in a manner that would affect the CSB. Onsite, runoff water from the southern portion of the CSB site flows in an easterly direction in a drainage swale that is adjacent to the patrol road slope along the south edge of the CSB site. This water flows in an easterly direction around the south end of the site, then north along the east side of the site to the 36-in. culvert that passes under the ramp road immediately northeast of the CSB. This culvert is the only water pathway that have any impact on the CSB. In an extreme case, the culvert could be plugged with debris at the time the PMP storm occurred and pond up to the lowest elevation of the road immediately north of the CSB. This road ramps from an elevation of approximately 705 ft next to the CSB to an elevation of approximately 707 ft at the road’s intersection with the access road coming in from the north. The rest of the site drains away from the CSB, which has a floor elevation of 709 ft. Additional rainfall would increase the depth of flow about 1 ft over the ramped access road to an elevation of 706 ft, but at no time would the water level build up to the 709-ft elevation that would threaten the CSB, affect its structural integrity, or allow water into the building. Calculations documented in CSB-PR-0001, *Vault Moisture – SS by RO*, of potential maximum rainwater entering the vault area show that a build-up of water inside of the vault is not a concern. Inside the vault, the evaporation rate will be much greater and the amount of condensation will be much lower than outdoors, even before the storage tubes are loaded with MCOs. After the storage tubes are loaded, the floor will be warm and any water that could enter will evaporate within hours.

**A2.3.3 Historical Summary of the Canister Storage Building Vault Structure**

The SNF Project CSB incorporates portions of the design and the completed below-grade sections of the canceled HWVP CSB. All of the designs and completed sections of the HWVP CSB were reviewed and met or exceeded the requirements for the SNF Project CSB. Portions of the vault that were completed prior to suspension of the HWVP project in 1993, including the base slab and portions of the walls, were reviewed, approved, accepted, and incorporated into the present design by the architect-engineer.

Future use of the CSB may include receiving and storing glass log canisters shipped inside of a large, heavy, shielded canister transporter. Approval of a separate safety analysis addendum will be required for any activities outside the scope of this FSAR.
A2.3.4 Canister Storage Building Shielding Design

The MCOs are received at the CSB via truck shipment within a transportation cask. The transportation cask design provides about 7.6 in. of steel for the side thickness, resulting in a worst case dose rate of about 50 mrem/h on contact at the mid-height cask surface according to CSB-SH-1002, Bulk Shielding for the Vault Release (Phase I) of the Canister Storage Building (CSB). This higher than acceptable dose rate for the CSB has resulted in additional shielding requirements for the CSB. The following paragraphs outline the shielding analysis performed for detailed design of the CSB.

The code selected to perform shielding calculations on the CSB vault was the personal computer (PC) version of MCNP, version 4A, developed and supported by the Los Alamos National Laboratory LA-12625, MCNP - A General Monte Carlo N-Particle Transport Code, Version 4A. This code was selected because the complex geometry of the source region, and the CSB could be modeled with a high degree of confidence.

Running the test problems supplied with the code packages has validated all codes used in shielding and criticality calculations at Fluor Daniel, Incorporated. The test problem results are shown to be in agreement either with the published documentation supplied as part of the code package, or with the sample problem output, if supplied. This information is documented, dated, and retained on file. Any changes, such as upgrades, corrections, and modifications, are incorporated into the documentation following rerunning the verification problems. MCNP, Version 4A, described in LA-12625, is a widely used and well accepted radiation transport code employed for a wide variety of radiation analyses including neutron, photon and electron transport, and criticality. Los Alamos National Laboratory has performed extensive calculations with MCNP, duplicating a wide range of experimental results, to validate the models contained in this code. The results of these benchmark cases are documented in LA-12196, MCNP: Photon Benchmark Problems, and LA-12212, MCNP: Neutron Benchmark Problems. Exercising the 25 sample problems supplied by Los Alamos National Laboratory tested the version of MCNP-4A used at Fluor Daniel, Incorporated, and the results were found to be in agreement within reasonable statistical limits. These sample problems are designed to exercise a broad range of the code’s computational capabilities.

Fractional densities for common materials, radiation sources, fluence rate to dose rate conversion factors, and guidelines for statistically adequate results used in the calculations are documented in CSB-SH-1002. Source terms used in the shielding calculations are referenced in CSB-SH-1001, SNF CSB Shielding Source Terms. Monte Carlo shielding analyses codes (e.g., the MCNP-4A) rely on sampling a large number of particles and tracing them throughout the modeled geometry. Because the results are a function of the number of initial source particles sampled and the number of particles reaching the specified dose points, the resulting mean “detected” dose rates have an implied statistical uncertainty (relative error) associated with them. Depending on the degree of attenuation and the complexity of the sample space, achieving unrealistically low relative errors can be extremely time-consuming (e.g., days of PC computing time) and expensive (additional labor time and potential schedule impacts).
In order to place practical limits on relative errors (a statistical term meaning 1-sigma deviation from the mean), guidelines of acceptable uncertainty ranges were established (CSB-SH-1002). These uncertainty ranges provided practical limits on the relative error for each estimated dose rate. Table 4 in CSB-SH-1002 presents interpretations of the relative errors with regard to the quality of the results.


The shielding calculations noted in this chapter and Chapter A4.0 include estimated dose rates for the vault intake stack and vault exhaust stack (exterior walls), the CSB operating area shelter exterior, the operating area floor, the interior vault walls, the exterior vault walls, the cask receiving pit, and the sampling/weld stations. Shielding calculations were updated for the same equipment if subsequent design changes appeared to have an impact on the calculated doses. The radiation doses cited in this document reference the specific shielding calculations performed to estimate the doses. The shielding calculations contain the assumptions, geometry models, and equipment dimensions used.

A2.4 FACILITY STRUCTURE

This section provides an overview of the facility and auxiliary structures including construction details such as basic floor plans, equipment layout, construction materials, controlling dimensions, and dimensions significant to the hazard and accident analysis activity. This information supports an overall understanding of the facility structures and the general arrangement of the facility as it pertains to hazard and accident analyses. The following subsections describe the below-grade reinforced concrete structures (vault, basemat, inlet plenum and outlet plenum), the at-grade reinforced concrete structures (operating deck and perimeter curb), the steel structures (storage tubes, storage tube covers, tube plug covers, and impact absorbers), and the CSB above-grade structures (intake structure, exhaust stack, operating area shelter, and support area building).
A2.4.1 Canister Storage Building Subsurface Structure

The below-grade reinforced concrete portion of the CSB (basemat embeds, vaults, inlet plenums, and exhaust plenums shown in Figures A2-5 and A2-6) is approximately 181 ft long by 166 ft wide by 48 ft deep. The floor area encompassing the load-in/load-out area, operating area, and sampling/weld area is approximately 230 ft long by 136 ft wide. The operating floor is enclosed by a steel-framed building, the operating area shelter. The reference elevation of the facilities at the top of the operating deck is 709 ft above mean sea level.

**Basemat.** The CSB basemat (foundation) is constructed at an elevation of 661 ft, 3 in. above mean sea level, or approximately 46 ft, 9 in. below the grade elevation of 708 ft. The basemat is nominally 5 ft, 6 in. thick; its surface elevation is 666 ft, 10.5 in. The distance from the basemat floor to the underside of the operating deck is 37 ft, 3 in.

**Vaults.** Only vault 1, the northernmost vault, has carbon steel storage tubes and an individual intake structure and exhaust stack in order to expedite the receipt of MCOs containing SNF. The exterior walls and air intake and exhaust plenums are 4 ft, 6 in. thick. Interior partition walls between vaults are 3 ft thick. The intake structures and exhaust stacks for the other two vaults, except the intake and exhaust stack bases, have not been constructed and are outside the scope of this FSAR.

The below-grade walls and basemat carry the loads associated with handling and transporting the MCO (i.e., transporter with cask, receiving crane, and MHM).

**Air Intake and Exhaust Plenums.** The air intake and exhaust plenums are concrete ducts providing for airflow into and out of the vaults. The CSB below-grade structure provides shielding for the air intake and exhaust plenums. The subsurface structure with 3-ft-thick interior walls provides shielding from the source term associated with SNF storage. The exterior walls of the subsurface structure were designed to meet the criteria given in 10 CFR 835 for uncontrolled access. Shielding calculations have assumed that each MCO contains the maximum activity value associated with Mark IV fuel listed in HNF-S-0425. The shielding calculations indicate that the dose rate at the intake stack base will be less than the design requirements (0.05 mrem/h [5.0 x 10^-7 mSv/h]) (CSB-SH-1002), of which less than 10% is attributed to neutron scatter with SNF as the source term. Heating of the concrete by the radiation field inside the vault has been calculated (7 x 10^-8 W/m²). The value is insignificant compared with convection heat transfer according to CSB-HV-0001, *A Canister Storage Building Vault Thermal Analysis*. Neutron and gamma effects from the spontaneous fission of californium were not included in the shielding calculations or concrete heating calculations because their contribution is negligible.

**Safety Considerations.** The CSB subsurface structure (including the basemat embeds, vault, air intake plenum, and exhaust plenum) is classified as safety class and important-to-safety Category A.
A2.4.2 At-Grade Reinforced Concrete Structures

The at-grade concrete structure includes the operating deck, the intake structure, the exhaust structure (Figure A2-6), and the perimeter curb.

The operating deck is a 5-ft-thick reinforced concrete structure (Figure A2-6) that forms the at-grade portion of the CSB. The operating deck is shown in plan view on Figure A2-7. The dimensions of the operating deck are approximately 230 ft north-south by 137 ft east-west. The operating area deck floor is bounded to the north by the load-in/load-out area (trailer vestibule and MCO service station) and support area building foundations, and to the south by the sampling/weld area foundation. The operating deck structure contains numerous through-thickness steel sleeves (embeds) that receive the storage tubes and tube plugs for both the standard storage tube and overpack storage tube locations in vault 1, and that provide a location for the tube plug cover plates in vault 1 or deck embed cover plates in vaults 2 and 3. These embeds are offset and arranged at a center-to-center distance of 4 ft, 8 in. east-west and 4 ft, 6 in. north-south, as shown in Figure A2-7. This distance between MCOs has been evaluated for criticality prevention and found to be adequate (see Chapter A6.0).

Two hundred twenty standard and six overpack embeds are provided in vault 1. Details of the standard tube, embed, and plug at the deck level are shown on Figure A2-8, sheet 1. Overpack embeds, tubes, and plugs are of similar dimensions with the overpack embed 6 in. larger in diameter (Figure A2-8, sheet 2).

Each vault is provided with penetrations through the operating deck to accommodate a rectangular intake structure 10 ft, 8 in. by 16 ft by 2 ft thick and a circular exhaust stack structure 11 ft, 6 in. in diameter and 4 ft, 6 in. thick. Each of the three exhaust stack bases is furnished with 2.25-in.-diameter anchor bolts to which the steel exhaust stacks can be fastened. The vault 1 intake and exhaust stacks are fastened using these anchor bolts. The intake structure and exhaust stack structures are designed for the NPHs listed in Table A1-1.

The sampling/weld area is an extension to the south of the operating deck. Its foundation is a reinforced concrete structure that houses process module mounting plates and seven process pits for sampling and welding (future) equipment. The dimensions of the sampling/weld area foundation are 35 ft, 3 in. by 138 ft, 11 in. by 5 ft thick; the reinforced concrete slab is supported at grade. A loading/staging area is located on the west side of the sampling/weld area for egress and ingress of equipment. The dimensions of the loading/staging area foundation are 24 ft, 11 in. by 27 ft, 4 in. An outer and an inner door in the loading/staging area serve as an airlock for area ventilation confinement.

The load-in/load-out area is an extension to the north of the operating deck. This area houses a cask receiving pit and provides trenches and anchor bolts for the rails serving the receiving crane, service station enclosure, and tent gantry hoist. The load-in/load-out area reinforced concrete slab is 138 ft, 4 in. by 29 ft, 9.5 in. by 5 ft thick supported at grade. A trailer vestibule forms an extension to the west of the load-in/load-out area. The dimensions of the
trailer vestibule slab are 42 ft, 7 in. by 50 ft, 9.5 in. by 5 ft thick. The reinforced concrete slab is supported at grade. An outer door and an inner door in the trailer vestibule serve as an airlock for operating area ventilation confinement. Two additional service pits are provided in the CSB load-in/load-out area. The easternmost pit is for MHM maintenance, and the center pit was designed for future transloading removal of Fast Flux Test Facility (FFTF) fuels from a transportation cask. Details of these service pits can be seen on Figure A2-9. A 10-ton screw jack assembly, mounted on the floor at the west end of the MHM maintenance pit, supports, elevates, and lowers the MHM mounting ring assembly and retractable nose assembly during maintenance operations.

Surrounding the operating area on the south, east, and west is a 4-ft, 4-in.-thick by 3-ft-high concrete perimeter curb. The curb is 2 ft wide and 8 ft high at the north wall of the load-in/load-out area for shielding support area building personnel from the MCO transportation cask. Doors are provided for personnel traffic through this shield wall. The operating area steel superstructure is attached to anchor bolts set at column locations throughout this perimeter curb. Survey markers (shown in Figure A2-3, sheet 1) are attached flush with the concrete deck and will be used in the deck settlement monitoring program (see Section A4.3.3).

**Safety Considerations:** The at-grade reinforced concrete deck structure (operating area deck [includes load-in/load-out area and sampling/weld area], intake structure, and exhaust structure) is classified as safety class and important-to-safety Category A.

### A2.4.3 Steel Structures

**A2.4.3.1 Storage Tubes.** Storage tubes (Figure A2-10) are located under the operating floor, accessed through the top of the operating deck, and supported from the basemat by a tube base assembly. The storage tubes are used to provide a controlled environment to vertically store MCOs. There are two types of tube assemblies. A standard storage tube assembly has been installed through each of the 220 standard embed openings in the operating deck. An overpack storage tube assembly has been installed through each of the six overpack embed openings in the operating deck. The overpack storage tube assemblies provide a sealed confinement function in addition to a structural function. The upper end of each tube has a bellows with an upper flange. The upper bellows flange of each tube assembly is seal-welded to the deck embed.

Each of the 220 standard storage tube assemblies in vault 1 includes impact absorbers, a tube base, and a tube plug. Each storage tube location will receive a plug cover plate. Embed covers will be placed at the floor level in the embed openings for vaults 2 and 3 until a future mission is identified that requires utilization of these openings.

The storage tube assemblies are centered at the top, welded to the deck embed, and restrained from lateral movement at the basemat by the tube base assemblies into which the tubes are inserted (Figure A2-11). The assemblies are affixed to plates embedded in the basemat slab and are designed with cooling passages to ensure that the temperature at the basemat slab will not
The standard and overpack storage tubes are provided with stainless steel expansion bellows to complete the assembly. The bellows assembly (Figure A2-12) permits thermal displacement of the storage tube, accommodates differential movement of the operating deck in relation to the basemat slab, and seals the operating area from the vault. The standard bellows assembly is 37 in. in diameter and 21 in. tall with a free length of 9.4 in. The bellows are constructed of 0.036-in.-thick stainless steel Type 304 series.

The standard storage tubes are 39-ft-, 7-in.-long, 0.5-in.-thick ASTM A36 carbon steel pipes, 28 in. in diameter, are closed at the bottom, and are designed to meet the requirements of the Boiler and Pressure Vessel Code, Section III (ASME 1995). The storage tube assembly is 41 ft, 4 in. long when the bellows are in place at the top. The overpack storage tubes are SA-671 carbon steel plate pipes of the same length and thickness as the standard storage tubes, and are designed to meet the requirements of the Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC, Article NC-6300 (ASME 1995), for air at a pressure of 100 lb/in² gauge. Overpack storage tube assemblies are provided in vault 1 for potential future use to store and monitor suspect MCOs.

A stress analysis of the standard storage tube documented in CSB-RM-0011, *ASME VIII Evaluation of Storage Tubes*, shows that it will withstand an internal pressure of 100 lb/in² gauge at 104 °C (220 °F). The normal pressure is 0 lb/in² gauge (vented through a breather filter to the operating area). There is sufficient clearance between the tube wall and impact absorbers to allow easy removal and installation and to prevent lateral loads from impacting the tube wall during staging or storage in the event of an MCO drop.

Standard storage tubes are provided with impact absorbers at the bottom of the tubes (bottom impact absorber) and between MCOs (intermediate impact absorber). The bottom impact absorbers limit the deceleration (maximum force allowed on the MCO is 679,000 pounds-force) of a maximum weight MCO dropped from the maximum height of 44 ft (35 g loading) and limit the radial load on the storage tube. An intermediate impact absorber placed on top of the bottom MCO is between the MCOs and prevents local damage to the canister cover assembly of the bottom MCO. The intermediate impact absorber sits on the flat surface of the bottom MCO canister cover assembly (Figure A2-2). The bottom and intermediate impact absorbers are designed to limit their sideward thrust on the storage tube during deformation to that of the tube working pressure. The lower and intermediate impact absorbers are shown on Figure A2-13. An intermediate impact absorber is put in place before placing a second MCO in a standard storage tube. Overpack storage tubes do not require an intermediate impact absorber because only one MCO will be stored and monitored in an overpack tube.
The lower and intermediate impact absorber assemblies (Figure A2-13) consist of a top flange plate, a top plate bolted to the flange plate, a base plate, and 3.5-in.-diameter energy absorbing tubes between the plates. In addition, the lower impact absorber has an intermediate plate separating the upper energy absorbing tubes from the lower energy absorbing tubes. The top, intermediate, and base plates have 0.25-in.-deep grooves providing alignment and a piloting surface for each tube component. Carbon steel was selected as the tubing material because of its ductility and energy absorbing capabilities. According to PacTec ED-037, CSB Impact Absorber Analysis Report, the following three impact absorber design features avoid the high initial short-duration force spike: (1) approximately half of the tubes are shorter by a one-half crush wavelength, (2) each tube has three dents, or crimps, separated by 120° of circumference, and (3) the impact absorber, after assembly, is precrushed. The shorter tubes are fitted with guide sleeves at the opposite end from the crimps. These guide sleeves serve to guide the “free” end of the shorter tubes into the grooves on the plate. The effect of these three design features is to minimize the high initial force that starts crushing of the tubes and to produce a smoother force-deflection curve for the impact absorber, effectively minimizing the deceleration forces on the MCO during a dropping accident. The impact absorber assemblies are held together by galvanized steel cables fitted around the circumference of the impact absorber and tensioned. After a potential crush event, the cables remain undamaged and hold the components together during removal of the damaged impact absorber. The number of tubes for each impact absorber, ranging from 13 to 17, will be based on full-scale testing of a predetermined quantity of impact absorbers using tubing from a batch of tubes provided by a single tube manufacturer. The testing results will determine the number of tubes to be used in each impact absorber assembly.

A2.4.3.2 Storage Tube Plugs. The standard tube plug assembly has a breather filter, provides radiation shielding, and allows access to the tube interior for gas monitoring through a quick-disconnect fitting. The overpack tube plug assembly provides radiation shielding; holds gas pressure in the overpack storage tube; and allows access to the tube interior for gas inerting and gas pressure monitoring. The standard and overpack tube plugs provide confinement of the MCOs. The tube plug has a cast steel top and concrete-filled lower portion. The barrel of the concrete tube plug is constructed of SA-36 carbon steel material. A 0.125-in.-diameter vent hole ensures that pressure inside the concrete barrel is relieved. The tube interior is accessed through two 2-in. holes drilled into the casting that connect to two 1.5-in.-diameter tubes.

The 45° sealing surface of the tube plug is mated to the upper 45° sealing surface of the storage tube. The seal is made using an O-ring seal (one O-ring seal for the standard tube plug and two O-rings for the overpack tube plug) constructed of ethylene propylene. The O-ring is compressed by the weight of the plug, and the 45° surfaces make metal-to-metal contact. The O-ring helps to prevent escape of potential contamination from the MCO surface into the operating area and also aids in keeping dust, debris, and liquid spills from leaking into the storage tubes from the operating area. The overpack sealing requirement is that the leakage be less than $1.3 \times 10^{-5} \text{ft}^3/\text{h}$ (at established tube pressure) to maintain an inert pressurized atmosphere around a suspect integrity MCO. In addition to the two O-ring seals, the overpack tube plug has a spider lockdown mechanism (Figure A2-14, sheet 2) clamping the overpack tube plug in place. The spider lockdown mechanism is designed to maintain the overpack tube pressure and prevent gas...
leakage from the overpack storage tube into the operating area. The double O-ring seals and spider lockdown mechanism provide confinement for potential radioactivity leaking from a suspect integrity MCO. There are no other design differences between the overpack storage tube plugs and the standard storage tube plugs (Figure A2-14).

Tube plugs are shown on Figure A2-14. The dose rate estimate, based on the Monte Carlo, three-dimensional geometry, MCNP-4A code streaming and scattering calculations (LA-12625), is estimated at 200 mrem/h (2.0 mSv/h) near the top of the plug-tube gap (top of the bottom straight plug section). The O-ring seal manufacturer data sheet, Selecting Elastomer Seals #23B (Parker 1989), shows that ethylene propylene retains its elastomeric properties at temperatures up to 250 °F and accumulated dose up to $10^7$ R. The maximum concrete temperature near the seals has been estimated to be about 115 °F (Figure A4-3). At a dose rate of 200 mrem/h, the accumulated dose for 40 years would be approximately $0.3 \times 10^7$ R. These values show that the tube plug seals should retain their sealing properties for the life of the facility. Sections A4.4.2 and A4.4.3 provide a more complete description of the storage tube plugs. At the surface of the operating deck, the upper limit of the dose rate caused by the radiation from the plug-tube gap is less than 0.2 mrem/h (2.0 x $10^3$ mSv/h). The radiation fields will be confirmed by routine monitoring by a health physics technician (HPT) during operations and filling of the vault. The analysis was performed on the concrete and steel composite plug. Details of the shielding analysis can be found in calculation CSB-SH-2002, Floor Plug/Deck Interface Analysis. Overpack storage tube plugs were similarly analyzed.

**A2.4.3.3 Tube Plug Covers.** Tube plug covers protect the tube plug connections and devices and fit flush with the operating deck. In vault 1 the tube plug covers are 3-in.-thick carbon steel disks provided with lifting lugs and a concentric inspection port. The deck embed covers in vaults 2 and 3 are 0.5-in.-thick carbon steel plates designed with tube steel reinforcement and with gaskets at deck embed contact surfaces. The deck embed covers provide for personnel safety and maintain a level floor surface. Both the tube plug and embed covers are shown on Figures A2-15 and A2-16.

**A2.4.3.4 Cask Receiving Impact Absorber.** An impact absorber is located in the cask receiving pit in the load-in/load-out area to elevate the transportation cask, absorb the impact of a dropped cask, and align it with the top of the operating deck. The cask receiving impact absorber assembly (Figure A2-9, sheet 5) consists of a top plate, a base weldment, 12 energy absorbing tubes between the top plate and base weldment, and an outer 0.5-in.-thick tube enclosure around the tubes. The top plate and base weldment top plate have 0.375-in.-deep grooves providing alignment and a piloting surface for each tube component. Carbon steel was selected for the tube material because of its ductility and energy absorbing capabilities. The three receiving pit impact absorber design features selected to minimize deceleration forces of an accidentally dropped cask-MCO are identical to the three design features selected for the bottom and intermediate impact absorbers and discussed in Section A2.4.3.1. Six of the crush tubes are shorter and symmetrically arranged, as shown in Figure A2-9, sheet 5. The four rods mounted circumferentially around the base weldment serve the same purpose as the galvanized cables for the bottom and intermediate impact absorbers. The base weldment is composed of three rings of
welded steel plate and pipes designed to support a cask–MCO placed in the cask receiving pit.

Steel cask guides (Figure A2-9, sheet 2) are located inside the cask receiving pit to align the cask with the shield hatch and MCO guide assembly (Figure A2-17).

**Safety Considerations.** The standard storage tube and tube base assembly are classified as safety class and important-to-safety Category A. The standard storage tube plug, lower flange, bottom impact absorber, and standard storage tube intermediate impact absorber are classified safety significant.

The overpack storage tubes and tube base assemblies are classified as safety class and important-to-safety Category A. The overpack storage tube plug, lower flange, and bottom impact absorber are classified safety significant.

The cask receiving impact absorber is classified as safety significant.

A2.4.4 Canister Storage Building Above-Grade Structures (Intake Structure, Exhaust Stack, Operating Area Shelter, and Support Area Building)

A2.4.4.1 Intake Structure and Exhaust Stack. The CSB intake structure is a 17 ft by 18 ft by 22 ft high steel structure mounted on top of a 15 ft by 20 ft by 57 ft high reinforced concrete tower that is constructed monolithically with the CSB vault. The steel portion of the intake structure is a combination of a braced tubular steel frame with steel shear panels at the base. The tower is analyzed and designed as an integral part of the CSB vault structure according to CSB-S-0028A, **Intake Tower Analysis**. The intake structure contains an inlet that increases static wind pressure by reducing the wind velocity at the windward side. A canopy is provided for increased wind pressure efficiencies.

The exhaust stack is a self-supporting 156-ft-, 10.5-in.-tall cantilevered structure. The diameter at the base of the stack is 13 ft, and the diameter at the top of the stack is 7 ft. The bottom section of the 13-ft-diameter stack base rises 55 ft and is constructed of 0.63-in.-thick carbon steel plate. The middle section continues upward 33 ft and tapers from 13 ft to 7 ft in diameter and also is constructed of 0.63-in.-thick carbon steel plate. The 7-ft-diameter top section of the stack continues upward another 57 ft and is constructed of 0.38-in.-thick carbon steel plate. The upper 44-ft section of the stack is provided with vertical wind spoilers that are each 7 in. tall by 11 ft long, made of 0.38-in.-thick carbon steel plate and designed to ASCE-7-93, **Minimum Design Loads for Building and Other Structures** guidance. Thermal gradients across the stack are of minimal design concern for the materials and temperatures involved. The stack is topped by a wind deflector tip as shown on Figure A2-1.

**Safety Considerations.** The intake structure and exhaust stack are classified as safety class and important-to-safety Category A.
A2.4.4.2 Operating Area Shelter. The operating area shelter is a steel-frame enclosure that covers the vault operating deck, the load-in/load-out area on the north, and the sampling/weld area on the south. The structure is approximately 230 ft long, 137 ft wide, and 55 ft high and sits on a 3-ft-high concrete curb around the vault deck. The structural system comprises load-carrying roof trusses spanning the east-west dimension and supported by columns spaced typically at 20 ft, 4 in. centers.

The trailer vestibule and loading and staging areas extend out from the western edge of the operations area and provide controlled transfer of equipment and materials into and out of the operating area shelter. The sampling/weld area opens directly into the southern end of the operations area and uses the same MHM.

The standing seam welded roof system provides an impervious weatherproof barrier over the entire CSB structure. The structural steel deck and roof framing system are designed to resist tornado wind loads and other NPHs as discussed in Section A4.4.1.4. One weather-sealed penetration, a locked roof hatch, through the roof provides for outside crane access to the operating area. Future use of the roof hatch is not part of the authorization basis and will require using the Unreviewed Safety Question process.

The exterior siding encloses the CSB structure in a continuous, airtight, weatherproof shell, except for the doors. All areas of the CSB have the same siding design, comprising a 3-in.-deep metal liner panel screwed to structural girt framing. The metal siding resists extreme wind-driven missiles and tornado wind loads and pressure drop as required by WHC-SD-SNF-DB-009. Fasteners will resist both pull-out and pull-over stresses induced by the specified wind load, as well as seismic and other forces (e.g., construction loads, roof loading, thermal changes).

The operating area shelter contains a 4-ft-wide egress passageway that generally follows the perimeter walls. This passageway connects all of the emergency exits. A passageway also separates the vault operations area from the sampling/weld area. Personnel working in the deck area can directly access the exits through the dedicated egress passageway. The waterproofed, sealed (using a resinous coating), exposed concrete floors and horizontal surfaces of the load-in/load-out area resist liquid intrusion, aid in preventing contamination of the concrete surfaces, and reduce future decontamination and decommissioning labor by minimizing contamination of the concrete.

Two radiation shield doors allow ingress to and egress from the support area building from the operating area shelter. A third shield door allows access from the operating area into the filter room. These doors were chosen to meet space limitations and provide shielding characteristics equivalent to a labyrinth design. Special exit-only doors, spaced around the exterior walls, provide the only emergency exit from the operating area shelter. Emergency lighting and local alarms are provided at each emergency exit door. The trailer vestibule and loading and staging area also have exit-only doors. The operating area shelter has eight personnel emergency exits. All of the operating area emergency exits open to outside of the building onto either concrete door landings, concrete sidewalks or a paved area (see Figure A2-3).
Personnel enter and exit the main corridor in the support area building through one personnel entry door. Shielding calculation CSB-SH-3004, *Dose Rates through North Wall*, determined the shielding required to keep the dose rate to personnel in the support area building below 0.2 mrem/h (2.0 x 10^{-3} mSv/h) using the MCO transportation cask as a source term. The dose to unshielded personnel from the MCO transportation cask was calculated at 3.8 mrem/h (3.8 x 10^{-2} mSv/h) at 66.5 in. The 2-ft-thick concrete north wall of the load-in/load-out area and the 2.5-in.-thick steel doors reduce support area personnel exposure from the MCO transportation cask during offloading. With such shielding, the highest dose rate expected in the support area building is about 0.026 mrem/h (2.6 x 10^{-3} mSv/h), which is well below the 0.2 mrem/h (2.0 x 10^{-3} mSv/h) desired for full-time occupancy.

A large (37 ft by 38 ft), fabric panel Megadoor\(^1\) acts as an inner airlock door large enough for the receiving crane to pass through when traveling between the trailer vestibule and cask receiving pit load-in/load-out area. A fabric-type curtain door was chosen because of the large opening size to be closed and the air sealing capability needed. An overhead metal telescoping door (24 ft by 20 ft) encloses the outer opening of the trailer vestibule at the northwest entrance to the CSB. Two more telescoping doors (14 ft by 14 ft) serve as inner and outer airlock doors for receiving equipment at the southwest entrance to the CSB. Telescoping doors have multiple steel panels that telescope into the door as the door is raised. The trailer vestibule and southwest entrance exterior telescoping doors are protected from tornado damage and accidental truck crashes by rolling gates (Figure A2-18). Each hydraulic-driven gate is positioned by a drive train in front of the telescoping doors when the telescoping doors are in closed position to minimize penetration during the design basis tornado (Table A1-1) or from a runaway vehicle. Warning indicators (rotating beacons and alarm bells), located on the west building exterior near the telescoping doors, are activated when the doors are moving. Each gate and its telescoping door are interlocked so that both operate simultaneously from a single set of push buttons. Safety edges on both the gates and telescoping doors, when activated, stop the motion of either the door or the gate. When the gate and the telescoping door have been in the open position for 15 minutes, an audible alarm activates and continues alarming for 60 minutes or until the door-gate “Close” push button is pushed. The gate has a manual operation feature that provides for operation of the door during loss of power.

Each exterior telescoping door is interlocked with the airlock door (Megadoor\(^\text{TM}\)) or the interior telescoping door to maintain a negative pressure inside the operating area. The Megadoor\(^\text{TM}\) is also interlocked with the receiving crane to prevent movement of the crane into the Megadoor\(^\text{TM}\) when the door is closed. Megadoor\(^\text{TM}\) interlock switches allow the receiving crane bridge to travel through the door opening only when the Megadoor\(^\text{TM}\) is in the full open position. The outside and inside doors are interlocked so that one door is always closed. Whenever one door is open, the other door is interlocked to prevent that door from opening until the first door is closed. This is to ensure a negative air pressure between the operating area (zone IIIa) and the outside air. Both trailer vestibule doors are operated using identical control.

\(^1\)Megadoor is a trademark of Megadoor Company.
systems. These doors are each controlled by a local control box. These controls initiate opening,
closing, and stopping of the door at any point in travel. Built-in adjustable limit controls are
provided to automatically disconnect power to the door operator control box at the full-open and
full-closed positions. Limit switches in the door header provide protection against damage in the
event of over-travel or a break in the lifting cable. This is used as a backup for the primary limit
control, which disconnects power from the operator control box. Each bottom door panel
contains various operating devices, including a pressure-sensitive bottom edge that automatically
reverses the door motion when an obstacle is met, and a special centering device that holds all
panels in proper lateral alignment to prevent side binding. A positive locking mechanism engages
automatically in both side guides when the door stops in the closed position and disengages
automatically when the opening action is initiated. An automatic load lock will hold the curtain in
a steady position whenever the curtain is stopped in vertical travel.

Hand-held fire extinguishers are provided in the operating area as defined by NFPA 10,
Portable Fire Extinguishers. A two-hour fire wall separates the operations area from the support
area building.

Seismic and tornado lateral forces are resisted through truss-column frame interaction in
the east-west direction and through a vertical bracing system along the east and west walls in the
north-south direction.

The CSB operating area shelter and supporting structures are protected from lightning
strikes in accordance with NFPA 780, Lightning Protection Systems. Air terminals are provided
along the perimeter of the operating area shelter roof. Section A2.9.3, “Lightning Protection,”
provides a description of the lightning strike protection features of the CSB operating area shelter.
Lightning arrestors are installed for 15 kV (nominal) systems. Lightning arrestors are spark-over
devices for overhead electrical distribution systems to protect their overhead electrical lines.
Grounding cables connect the lightning arrestors and the operating area shelter air terminals to the
CSB ground grid, an underground wire cable system that protects the CSB structures from
electrical hazards.

Safety Considerations. The operating area shelter is classified as safety significant and
important-to-safety Category B. The rolling shield gates are safety significant. The Megadoor™
and the telescoping doors are classified as general service.

A2.4.4.3 Support Area Building. The building that houses the CSB support area is a single-
story, steel-frame structure approximately 150 ft long, 53 ft wide, and 16 ft high. The support
area building is north of the CSB vaults and operating area shelter at the same 709-ft, 0-in.
elevation as the operating deck. The concrete slab is 8 in. thick constructed at grade. Below-
grade reinforced concrete footings are placed at equipment and column locations.

The support area building abuts the north edge of the operating area shelter and contains
several rooms: equipment spaces (including heating, ventilation, and air conditioning [HVAC],
filters; and instrument air); electrical room; two empty utility rooms with space heaters; control
room and security control space; HPT office; count room; regulated change room; operations office; and airlocks. The control room is essentially a monitoring station rather than a control room as defined by DOE conduct of operations standards. The equipment areas have dedicated access doors for equipment access and servicing. Exits from these areas are through the main corridor or, in the electrical and empty utility rooms, directly to the outside.

The support area building has a shed roof sloping down from the common wall adjoining the operating area. A structural steel deck is attached to the top of the roof-framing system. The support area building roof has an occasional-use access ladder, welded to brackets, that extends up the outside of the building, providing access to the ventilation exhaust stack and effluent-monitoring system probes.

Exterior siding and roofing for the support area building is of equal grade and finish appearance to the operating area shelter. The siding and exterior wall design for the support area building, including exterior joints and interface points, also is the same as described for the operating area shelter. A relatively small telescoping door on the west side (10 ft by 10 ft) opens to receive supplies into and discharge waste from the support area building. Warning indicators (rotating beacon and alarm bell) located on the west building exterior near the telescoping door are activated when the door is moving.

The primary personnel entrance for the CSB is at the northeast corner of the support area building near the HPT office, control room, and operations office. A main corridor crosses the entire support area building between the primary personnel entrance on the east and the loading dock on the west. This corridor provides an exit path to the primary support area building exits. Operating personnel enter the primary personnel entrance, don protective clothing in the change room, and enter the operating area shelter from an airlock in the support area building. The airlock ensures HVAC flow is from the support area building to the operating area. Personnel return from the operating area shelter by a similar path, passing through a portal contamination monitor, for clothing removal, decontamination, and egress. Personnel may access the HPT and operations offices, change and control rooms, and various other rooms through 12 single-wide interior personnel doors and 1 double-wide interior pair of doors. All doors (airlock and personnel) have air seals around the jambs. They are rated for at least 20 lb/ft² pressure and for 30 lb/ft² in the operating area. The HPT office has a view window to the personnel decontamination area.

Acoustical features provided for the control room in the support area include a suspended acoustical tile ceiling and sound board, which is installed on separate metal stud-framed wall partitions on both sides of the wall adjoining the empty utility room. The acoustical features are provided to comply with the requirements of DOE Order 6430.1A, Section 0950, “Acoustical Treatment.”

The personnel monitoring and decontamination room, count room, step-off pad, and regulated change room areas have a depressed floor level with an access floor aligned with the surrounding finished floor. The flooring in these access floor areas has perforations to allow
sprinkler discharge to accumulate in the depressed areas until the discharge can be
decontaminated and removed. Equipment areas have waterproofed, sealed, exposed concrete
floors and waterproofed, sealed, horizontal surfaces of the equipment pads. The sealed surfaces
resist liquid intrusion, aid in preventing contamination of the concrete surfaces, and reduce future
decontamination and decommissioning labor by minimizing contamination of the concrete. Sumps
are provided in the utility rooms, the HVAC equipment room, and mechanical equipment rooms.
Curbs and exterior access-only doors to the utility rooms ensure that liquids (fire sprinkler
actuation) will collect and drain away from the support area building.

Entry to the electrical room is through one tall exterior double door (6 ft by 9 ft) with a
removable top transom that allows equipment installation and servicing. Two similar, though
shorter, standard height (6 ft by 7 ft) pairs of exterior service doors provide access to the two
utility rooms. The utility rooms also have wall louvers. The room containing the continuous
airborne effluent monitor (CAEM) and the safeguards and security equipment is accessible from
inside.

The foundations and superstructure of the support area building are structurally isolated
from the CSB vaults and operating shelter. The building houses equipment for electrical, HVAC,
CAEMs, facility operations monitoring, and change rooms. A 28-in.-diameter, 75-ft-high steel
HVAC exhaust stack, supported at grade and guyed at the roof, penetrates through the building
roof. The structural system comprises roof beams that span the east-west dimension and are
supported by girders and columns. Seismic and tornado lateral force resistance is achieved by
girder-column frame interaction in the north-south direction and through a vertical bracing system
along the north and south walls. The building is brace-framed in the east-west direction. The
foundations for the building and equipment are spread footings located below the frost depth
according to CSB-S-0019, SNF Canister Storage Support Building.

Air terminals installed around the perimeter of the support area building roof provide
protection from lightning strikes in accordance with NFPA 780. The HVAC stack is provided
with two exposed cables located on opposite sides of the stack about 3 ft above the concrete
base. Grounding cables connect the support area building air terminals and HVAC exhaust stack
to the CSB ground grid, an underground wire cable system for protecting the CSB and its support
structures from electrical hazards. Section A2.9.3 provides a description of the lightning strike
protection features of the CSB support area building and HVAC exhaust stack.

Safety Considerations. The support area building is classified as general service, except the
foundation, which is classified as safety significant.

A2.5 PROCESS DESCRIPTION

Individual operations within the CSB are described in this section. The three major
operations are baseline operations, passive cooling, and overpack storage tube purge operations.
The details include basic parameters; summaries of types and quantities of hazardous materials;
equipment; instrumentation and control systems and equipment; basic flow diagrams; and
operational considerations of the individual processes and the entire facility, including major interfaces and relationships between SSCs. The intent of this information is to provide an understanding of the assessment of normal operations and insight into the types of operations for which a safety management program must be planned.

A2.5.1 Baseline Operations Involving Handling of Multi-Canister Overpacks

Baseline operations at the CSB include receiving operations, cask servicing operations, MCO operations, and MCO sampling operations. All operations in this subsection involve safe handling of sealed MCOs containing SNF. During normal operations, a maximum number of four MCOs would be in process in the operating area:

1. One MCO in the load-in/load-out area being prepared for receipt and transport by the MHM

2. One MCO in sampling/weld pit 2

3. One MCO in sampling/weld pit 7

4. One MCO being transported by the MHM.

A2.5.1.1 Receiving Operations. An MCO, loaded with SNF, is shipped from the CVDF to the CSB trailer vestibule (Figure A2-9) inside a transportation cask. The transportation cask is unloaded from the trailer, carried to the cask receiving pit, and preparations are made for cask servicing operations. The following sections describe the equipment and operations involved in receiving a cask and preparing for cask monitoring.

A2.5.1.1.1 Cask Receiving. The receiving crane is designed to offload an MCO transportation cask from a transport trailer and transfer the cask to the cask receiving pit. The receiving crane and appurtenances offload the MCO transportation cask from the transport trailer. After an MCO has been removed from the cask by the MHM, the receiving crane retrieves the cask, including an empty MCO, and places the cask back on the transport trailer.

Major Components. The following major components are used during the offloading of a transportation cask:

- Receiving crane
- Cask lifting yoke
- Rail frog assemblies
- Cask receiving impact absorber.

Equipment Description and Operational Considerations. The receiving crane is a 60-ton-capacity gantry crane with one 60-ton main and one 10-ton auxiliary wire rope hoist.
supported on a top-running trolley (Figure A2-19). The crane operates in a clean indoor
environment in the trailer vestibule and load-in/load-out areas of the CSB. The crane is equipped
with a shielded operator station and remote accessories, such as a powered rotating hook, local
control panels, and a radio-controlled operator station.

HNF-SD-TP-SARP-017, Safety Analysis Report for Packaging, Onsite, Multi-Canister
Overpack Cask, provides the safety basis for the cask–MCO during transportation activities. This
FSAR provides the safety basis for the cask–MCO during CSB activities after arrival of the
cask–MCO at the CSB. A cask transported to the CSB enters the trailer vestibule area on a
transport trailer (Figure A2-20). Trailer tire stops installed on the floor of the trailer vestibule
area prevent the transport trailer from backing too far. The overall length of the tractor-trailer is
approximately 54 ft, and the trailer length is approximately 40 ft. After the trailer is backed into
the trailer vestibule, the tractor leaves the area and the telescoping door is lowered. The cask
support and tie-down system hold the cask in an upright position surrounded on all sides by a
work platform. A stairway provides access for operations personnel to the elevated platform for
cASK inspection, radioactivity monitoring, loosening the cask tie-downs, and latching and
unlatching the cask-lifting yoke (Figure A2-21) and crane hook. The cask-lifting yoke has two
hooks used for lifting the cask (see Figure A2-21). The cask-lifting yoke and the receiving crane
main hook maintain the lifting height of the cask–MCO below 60 in. The receiving crane main
hook engages with the cask lifting yoke to lift the cask.

Trucks carrying empty MCOs enter the trailer vestibule and the empty MCOs are off-
loaded onto empty MCO transport dollies (Figure A2-22). The empty MCO cart is used to move
the empty MCO to the east end of the load-in/load-out area for storage until the empty MCO is
placed in the transportation cask.

Air/fuel dams (Figure A2-9, sheet 6) in the receiving crane rails, under the Megadoor™,
minimize air leakage paths so that the operating area HVAC can maintain a negative pressure.
The air/fuel dams also provide blockage against burning diesel fuel from entering the operating
area during the postulated fuel spill and fire accident in the trailer vestibule area.

The receiving crane gantry has a span of 27 ft, 6 in. and can travel 161 ft east-west. The
crane travels on runway rails that are recessed and flush with the operating floor. The runway
travel of the receiving crane intersects that of the MHM, which is also a gantry crane system. The
intersecting runway rails are coplanar and therefore have been provided with rail frogs that match
the bridge wheels on both cranes. The rail frog, with stop assembly (Figure A2-9), has been
designed with a rail stop and travel limit switches that prevent the cranes from colliding. An
emergency stop push button installed on the north wall, next to the receiving crane travel limit
floor marking, can be pushed by an individual to stop the receiving crane and prevent receiving
crane travel over the FFTF pit or the MHM maintenance pit. This push button is connected in
series with the shunt contact in the seismic trip panel and stops receiving crane travel when
pushed. In addition, solenoid-powered switch strikers (SPSSs, Figure A2-23, sheet 3) prevent the
receiving crane and the MHM from simultaneously occupying the overlap zone of the service
area. The SPSSs prevent the MHM and the receiving crane from colliding. This is particularly
important when the receiving crane is lowering an MCO cask into the cask receiving pit. As the receiving crane enters the overlap area where the MHM can collide with it, the receiving crane toggles two roller arm limit switches (RALs) that remove power from two SPSSs mounted on the east concrete curb of the CSB (Figure A2-23, sheet 4). The SPSSs are spring loaded and rotate striker arms to a position that will toggle the north slow and north stop limit switches (Figure A2-23, sheet 2) on the MHM. Actuation of the MHM limit switches, if the MHM attempts to enter the overlap zone, removes power from the MHM bridge drive motors and the MHM is prevented from entering the overlap zone. Similar MHM RALs and SPSSs for the receiving crane, activated by the MHM as it enters the load-in/load-out area, cut off power to the receiving crane if it approaches the load-in/load-out area, preventing the receiving crane from colliding with the MHM.

Jib cranes are provided for repositioning the rail frogs after one crane leaves the load-in/load-out area and before the other crane enters the load-in/load-out area. The rail frogs are expected to be repositioned twice for each MCO received. The south jib crane (Figure A2-9), equipped with a hoist and lifting device, has been designed to lift and rotate the rail frog and stop. The north jib crane (Figure A2-9), also equipped with a hoist and lifting device, has been designed to lift and rotate the north rail frog. Both jib cranes will be manually rotated into position as needed. When not in use, each jib crane will be secured in its storage position to prevent a collision with either of the cranes.

The limits of receiving crane gantry travel are established by bridge stops in the east and west building walls. These stops are designed to be contacted by bumpers mounted on the gantry trucks to absorb the energy of a kinetic impact. A positioning/interlock control system on the receiving crane indicates the location of the crane within any 6-in. increment of total crane travel. The resolver and limit switch dual channel interlock system limits crane travel over the FFTF pit and MHM maintenance pit. At a predetermined distance from the FFTF pit, the resolver controls will automatically stop movement of the receiving crane. A second function, controlled by a limit switch striker system mounted on the bus bar, prevents receiving crane travel over the FFTF pit or the MHM maintenance pit. If it is determined that the receiving crane is not carrying a cask, two different actions are required for the receiving crane to travel over the FFTF pit. The first action requires use of a control switch on the receiving crane control panel, and the second action requires personnel to use a supervisor-controlled fortress key (located remote from the receiving crane control panel). With the receiving crane emergency push button in the enable position and the control switches in the override mode, the receiving crane is permitted to travel over the FFTF pit. An interlock system on the crane prevents the crane from traveling east of the FFTF pit when the load suspended from the main hoist is identified as a transportation cask containing an MCO. These interlocks prevent the crane from hoisting the cask-MCO above the FFTF or MHM maintenance pits. A drop of the cask-MCO into either of these pits is prevented by an operator activating the receiving crane emergency stop push button and the receiving crane control interlocks.

Amber hazard lights mounted near the three doors providing entrance and exit from the support area building to the operating area (doors 008, 013, and 022) warn operating personnel
that the receiving crane is in the immediate vicinity. The lights (ceiling-mounted for doors 013
and 022 and wall-mounted for door 008) illuminate and shine on the doors whenever the receiving
crane is close to the door, and they remain lit until the receiving crane moves away from the
defined area close to the door. The hazard lights are a worker safety feature warning personnel to
be especially careful and to avoid injuries that might be caused by contact with the receiving
crane.

The receiving crane has a hook coverage capability that extends beyond the CSB SNF
requirements. The maximum hook elevation of 27 ft for the main hoist hook is needed to lift a
transportation cask containing FFTF fuel from the road truck. This hook elevation is
approximately 5 ft higher than the SNF cask transportation requirements. The main hoist of the
receiving crane is provided with height-limiting devices. These devices are a rotary hoist travel
limit switch and a weighted, lever-operated, up-limit switch. These devices are mounted on the
receiving crane's trolley floor. The switches are normally in closed position and in series with
other crane protective devices to provide a signal to the hoist-adjustable frequency drive. These
switches are activated when the receiving crane's main hoist hook block rises above the preset
height. When activated, the switches removed power from the main hoist motor and stop travel
of the hoist. These switches provide defense-in-depth safety features to prevent two-blocking the
hoist. The height of the cask lifting yoke is the safety feature for limiting the cask lift height to
60 in.

The receiving crane is expected to receive some radiation exposure when the transportation
cask is being moved. Based on an analysis of the side exposure to the receiving crane
documented in CAL-ME-007, Radiation Shield Requirements — Westinghouse Hanford
Company Receiving Crane, a maximum rate of about 11 mrem/h (0.11 mSv/h) is obtained (about
4 x 10^3 rad for a 40-year lifetime total integrated dose). This rate is a conservative value because
it is unlikely that the receiving crane will experience a 40-year continuous work schedule.
Therefore, the crane is expected to operate for the 40-year design life without significant
deterioration from radiation exposure.

The receiving crane has a shielded operator station mounted on the north bridge truck.
The shielding is 3.5-in.-thick carbon steel with viewing windows that have a shielding equivalency
of 3.5 in. of iron or 1 MeV photons reduced by a factor of 50. Shielding calculations performed
by the receiving crane supplier confirm that a shield thickness of 3.5 in. of steel results in an
individual dose rate of ≤0.2 mrem/h (≤2.0 x 10^3 mSv/h) (CAL-ME-007). In addition, the
specification requirement that 1 MeV gamma photons be reduced by at least a factor of 50 is met
by the lead glass shielding window when using a source based on 270 Mark IV fuel elements.
This source is more representative of the actual source to which personnel in the cab will be
exposed. The contact dose rate on the receiving crane is less than the specification requirement of
11 mrem/h (0.11 mSv/h). Dose measurements and shielding equivalency estimates will be
performed during startup and routine operations.

The receiving crane, including its load blocks, will not be exposed to temperatures,
pressures, or chemicals that would require special design considerations. The receiving crane
design provides for fail-safe features (e.g., load blocking and brakes) to be applied upon loss of power. Also, when an earthquake is detected, wall-mounted accelerometer channels in the northeast corner of the load-in/load-out area automatically disconnect the power source for the receiving crane. The seismic power disconnect eliminates potential failures resulting from seismic-initiated failures of the electrical interlock control system, which is not seismically qualified. The trip circuit is located in the power supply to the receiving crane and includes three tri-axial accelerometer seismic trip system channels connected to redundant power contactors in series. Any of nine sensors can cause both contactors to open. If only one contactor is functioning, the power circuit to the receiving crane will open. According to Report QA8685, *Commercial Grade Item Dedication*, the seismic trip system setting is 0.19 g peak ground acceleration based on guidance in ASME NOG-1-1995, *Rules for the Construction of Overhead and Gantry Cranes*, that the setpoint include the maximum horizontal and vertical spectra and 90% MHM weight participation at 7% damping. The estimated horizontal seismic spectral response acceleration values for CSB buildings (5% damping) and cranes (7% damping) are approximately two times the 0.35 g peak ground acceleration horizontal value. The trip setting, therefore, is approximately 54% of the building and crane response acceleration values to the design basis earthquake. This setpoint trips power to the receiving crane to protect the MCO from damage during the postulated design basis earthquake. The panels containing the accelerometers, signal conditioners, and trip units are rigidly mounted and seismically qualified.

The impact absorber in the cask receiving pit (Figure 2-9, sheet 5) is designed to reduce the likelihood of damage to an MCO inside the cask if the cask is accidentally dropped by the receiving crane.

**Safety Considerations.** The MHM rails and rail frogs, the MCO, and the transportation cask are classified as safety class and important-to-safety Category A. The receiving crane structure and hoist are classified as safety significant and important-to-safety Category B. The transportation cask shielding, the cask receiving impact absorber, the cask lifting yoke, and the MCO shield plug shielding are classified as safety-significant. The receiving crane positioning/interlock control system, the MHM collision avoidance system, and the receiving crane lift height limit switches are classified general service.

**A2.5.1.1.2 Preparation of Cask for Servicing.** The containment tent, tent gantry hoist (CRN-008), and auxiliary equipment are designed for preparing the cask for servicing operations and, after cask servicing operations (see Section A2.5.1.2), for safe handling of the transportation cask lid, and for preparing the cask for removal of the MCO by the MHM.

**Major Components.** The following major components are included in the process of preparing a cask for servicing:

- Cask receiving pit
- Tent gantry hoist
- Shield hatch and MCO guide assembly
- Cask lid automated torque tool
- Containment tent.
Equipment Description and Operational Considerations. The 5-ton motorized tent gantry hoist (also provided with manual hoist raise and lower capability) is used to remove the cask lid in the cask receiving pit and replace it after cask servicing operations are completed. The crane has an underhung electrical wire rope hoist. The crane does not exceed 8 ft in total height and has sufficient hoist capacity to effect a maximum lift of 5 ft. The underhung electrical wire rope hoist is equipped with brakes, end stops, and a free rotating hook. The wheels of the tent gantry hoist are designed to withstand a minimum loading of 5,000 lb each and are provided with locking devices to prevent movement while in use. The tent gantry hoist has an acceleration control, and its traveling speed does not exceed 35 ft/min. The wire rope hoist has a lift speed of 10 ft/min maximum, and the motorized trolley has a 25 ft/min maximum speed. After a power failure, the cask lid may be manually lowered.

The cask lid automated torque tool (Figure A2-9, sheet 7) provides automatic loosening, tightening, and torquing of the cask lid bolts. This tool uses pneumatically controlled torque wrenches mounted to a motor-driven, programmable-logic controlled, turntable fixture to torque the transportation cask bolts to prespecified torque values in a required bolting sequence. The cask lid automated torque tool has five main subassemblies: (1) frame and turntable assembly, (2) drive assembly, (3) pneumatic system, (4) control system, and (5) facility utilities. The frame and turntable assembly is self-aligning with the cask by means of guides that align to the lifting trunnions and tapered alignment pins on the cask lid. Personnel connect the tent gantry hoist hook to the frame and turntable assembly lifting yoke to lift the torque tool and places it on the transportation cask lid, aligning the guides with the tapered alignment pins. After disconnecting the tent gantry hoist lifting hook the following facility utilities are attached to connectors mounted on the lifting yoke: (1) an air supply, (2) a power cable, and (3) an umbilical cord (containing a power supply cable, an air line, and an interface cable) from the automatic torque wrench operator control panel. The drive assembly provides rotary motion for the turntable by means of an AC gear motor coupled through a 24 VDC electric clutch-brake to a chain drive and sprocket. The pneumatic system drives rotary operation of the torque wrenches, the Z-axis torque-wrench positioning system, and the index alignment pins. Personnel use the programmable-logic-control system to loosen the cask lid bolts (two at a time) for removal of the cask lid by the tent gantry hoist. After removing the MCO and inserting an empty MCO, the cask lid automated torque tool is again placed on the cask lid and personnel use the control system to tighten the cask lid bolts using the bolting sequence and prespecified torque value.

After cask servicing operations have been performed and the cask lid bolts have been loosened by the cask lid automated torque tool, the tent gantry hoist removes the cask lid, stores it on the cask lid stand, positions the MCO guide on the cask, and positions the shield hatch ring and shield hatch plate (shield hatch and MCO guide assembly, Figure A2-17) over the cask receiving pit. The 8-in.-thick MCO guide and the 10-in.-thick carbon steel shield hatch ring and shield hatch plate replace the cask lid function of shielding the MCO during MCO removal by the MHM. The radiation dose rate at the shield hatch and MCO guide assembly surface has been calculated at less than 0.5 mrem/h (5 x 10^{-3} mSv/h) in CSB-SH-3001, Service Pit Dose Rates. This hatch has double seals that mate to the MHM. The shield hatch and MCO guide assembly form an inverted cone opening, which mitigates accidental dropping of an MCO into the cask.
A containment tent has been designed to be used during off-normal operations to minimize potential contamination spread during servicing of a pressurized transportation cask received at the CSB. The containment tent, if needed, is a welded, carbon steel frame structure (material conforming to ASTM A36/A36M, Standard Specification for Carbon Structural Steel) with an inner attached flexible plastic polyvinyl chloride tent. The steel structure is 12 ft by 8 ft by 7 ft tall. The structure's height provides MHM bridge clearance (approximately 2 ft, 10 in.) when the MHM travels to the north end of the operating area. The containment tent is staged outside of the trailer vestibule area and is brought into the load-in/load-out area by a forklift and the receiving crane. If needed, the containment tent is placed over the cask receiving pit after a transportation cask loaded with an MCO is placed in the pit and cask servicing operations have determined that the cask is pressurized. The containment tent consists of three rooms separated by doors to provide entrance and egress of personnel while maintaining contamination control of the immediate area around the cask receiving pit. The primary purpose of the containment tent is to contain any contamination that may escape during cask sampling activities. A ceiling access sleeve is provided that allows for lowering a crane hook to lift the cask lid if required. The containment tent is removed from the cask receiving pit after cask servicing operations.

The containment tent is equipped with a high-efficiency particulate air (HEPA)-filtered exhaust unit (EF-005) with a built-in standby exhaust fan. One fan is in standby while the other is in operation. After the containment tent is moved into position, the flexible power cord for the containment tent exhausters is plugged into a power receptacle in the cask receiving service pit. The cask receiving service pit also provides helium and compressed air supply lines and connectors for use at the cask receiving pit. The exhaust system will maintain a negative pressure between the inside and the outside of the containment tent at a 1,000 ft³/min flow rate during the servicing operations. The exhaust unit discharges into the operating area where the air combines with the building ventilation system and discharges to the exhaust stack. On a loss of normal power, the containment tent exhaust system shuts down until power is restored.

The containment tent and the filtered exhaust system, when used, form the confinement envelope for the ventilation area around the cask receiving pit. To achieve and maintain a negative pressure the enclosure is sealed to the floor, and the openings and joints in the material of the enclosure are designed to efficiently maintain the seal.

The containment tent is made of a fire-retardant nuclear-grade polyvinyl chloride fabric in accordance with Chapter 12 of NFPA 701, Standard Methods of Fire Tests for Flame-Resistant Textiles and Films, and has a maximum tearing strength that is in accordance with ASTM D 2261, Standard Test Method for Tearing Strength of Woven Fabrics by the Tongue (Single Rip) Method (Constant-Rate-of-Extension Tensile Testing Machine); and ASTM D 2262, Standard Test Method for Tearing Strength of Woven Fabrics by the Tongue (Single Rip) Method (Constant-Rate-of-Traverse Tensile Testing Machine). The tent is large enough to house the cask lid and the shield hatch and MCO guide assembly. The tent is equipped with service sleeves, 3-in.-diameter minimum, to bring services into the enclosure as needed to support ventilation, to supply power, and to introduce tools.
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HNF-SD-SNF-SARR-005 specifies a maximum time of 135 days that an MCO can remain sealed inside the transportation cask without venting after vacuum drying operations at the CVDF. The MCO should be removed from the transportation cask within this time limit or the cask–MCO annulus vented to prevent the potential for a flammable gas buildup in the cask cavity (HNF-SD-SNF-SARR-005).

Table A2-1 lists the operating conditions for which the cask receiving system has been designed.

Safety Considerations. The shield hatch and MCO guide assembly are classified as safety significant. The cask receiving pit is classified as safety significant. The containment tent, tent gantry hoist, and appurtenances are classified general service. The tent gantry hoist is important-to-safety Category C because it could potentially drop the cask lid causing minor damage to the top of an MCO.

A2.5.1.2 Cask Servicing Operations. The transportation cask servicing system is designed for checking the pressure of the transportation cask and, if necessary for recovery operations, taking a sample of the gases inside the cask and purging the gases from the cask. The cask servicing system can be used by personnel to check the pressure of the received cask. Gases inside the cask are purged or vented with helium to ensure that potential hydrogen concentrations are diluted below flammable concentrations.

Major Components. The following major components are part of the cask servicing operations:

- MCO servicing HEPA filter
- Flexible steel hose
- MCO servicing instruments
- Sampling equipment.

Equipment Description and Operational Considerations. Personnel connect the cask servicing system (Figures A2-24 and A2-25) flexible steel hose to the cask access port on the cask lid. The servicing system pressure gauge indicates the internal pressure of the cask space surrounding the MCO. A pressure increase up to 3 lb/in² gauge is possible in the transportation cask because the MCO and transportation cask may increase in temperature after the cask lid is bolted on and the transportation cask is transported to the CSB. The temperature increase causes the air pressure inside the transportation cask to increase. A cask pressure less than 3 lb/in² gauge indicates that the MCO can be handled using normal procedures. The normal operating steps include purging or venting gases from the cask through the cask servicing system's HEPA filter and into the operating area HEPA filter housing. These gases are mixed with helium to ensure that potential hydrogen concentrations that may be present are diluted to below flammable concentrations. The cask bolts are removed, cask lid is removed, and the shield hatch and MCO guide assembly is placed in the cask receiving pit opening using the tent gantry hoist. These steps prepare the MCO for lifting by the MHM.
<table>
<thead>
<tr>
<th>Component</th>
<th>System design conditions</th>
<th>System operating conditions</th>
<th>Operating environment</th>
<th>Design operating environment</th>
<th>Material</th>
<th>Material</th>
</tr>
</thead>
<tbody>
<tr>
<td>Containment tent</td>
<td>60°F to 104°F, 0% to 10%</td>
<td>60°F to 85°F</td>
<td>60°F to 85°F</td>
<td>60°F to 104°F</td>
<td>Galvanized steel, ASTM A653b</td>
<td>Galvanized steel, ASTM A653b</td>
</tr>
<tr>
<td>Service station exhaust</td>
<td>60°F to 104°F, 0% to 10%</td>
<td>60°F to 85°F</td>
<td>60°F to 85°F</td>
<td>60°F to 104°F</td>
<td>Galvanized steel, 304 stainless</td>
<td>Galvanized steel, 304 stainless</td>
</tr>
<tr>
<td>Receiving crane</td>
<td>60°F to 104°F, 0% to 10%</td>
<td>60°F to 85°F</td>
<td>60°F to 85°F</td>
<td>60°F to 104°F</td>
<td>Manufacturer standard</td>
<td>Manufacturer standard</td>
</tr>
<tr>
<td>Test gatry box</td>
<td>60°F to 104°F, 0% to 10%</td>
<td>60°F to 85°F</td>
<td>60°F to 85°F</td>
<td>60°F to 104°F</td>
<td>Carbon steel</td>
<td>Carbon steel</td>
</tr>
<tr>
<td>Cask filling yoke</td>
<td>60°F to 104°F, 0% to 10%</td>
<td>60°F to 85°F</td>
<td>60°F to 85°F</td>
<td>60°F to 104°F</td>
<td>Stainless steel and carbon</td>
<td>Stainless steel and carbon</td>
</tr>
<tr>
<td>Cask lid automated torque</td>
<td>60°F to 104°F, 0% to 10%</td>
<td>60°F to 85°F</td>
<td>60°F to 85°F</td>
<td>60°F to 104°F</td>
<td>Carbon steel, ASTM A242</td>
<td>Carbon steel, ASTM A242</td>
</tr>
<tr>
<td>Shield bracket and MCO guide</td>
<td>60°F to 104°F, 0% to 10%</td>
<td>60°F to 85°F</td>
<td>60°F to 85°F</td>
<td>60°F to 104°F</td>
<td>NA</td>
<td>NA</td>
</tr>
<tr>
<td>Shield bracket and MCO guide assembly</td>
<td>60°F to 104°F, 0% to 10%</td>
<td>60°F to 85°F</td>
<td>60°F to 85°F</td>
<td>60°F to 104°F</td>
<td>NA</td>
<td>NA</td>
</tr>
</tbody>
</table>

This table is for information only and does not provide operational controls.

- MCO = multi-cask overpack.
- NA = not available.

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A pressure ≥3 lb/in² indicates that the MCO seals may have released gases into the cask and is considered an off-normal condition according to HNF-4509, *Evaluation of Canister Storage Building Cask Receipt Pressure*. If pressure in the cask is higher than 3 lb/in² gauge, the cask servicing system is designed for taking a sample of the gases, purging gas from the cask, and replacing it with an inert gas. This allows the cask lid to be removed to prepare the suspect MCO for placement in an overpack tube for monitoring (Section A2.5.3 summarizes overpack storage tube monitoring operations). The cask servicing system provides HEPA-filtered pathways to achieve the safety function of preventing or mitigating radiological releases during cask purging.

The cask servicing system provides inert gas to ensure that cask servicing activities prevent gases inside the cask from becoming flammable or explosive. Excluding oxygen from the cask, taking a gas sample from the cask, results of laboratory analysis of the sample for hydrogen, and purging hydrogen to safe concentration levels before release through the ventilation building system provide assurance that safe operating conditions are being maintained.

Venting of the internal cask atmosphere occurs through HEPA filter FH-2 into the load-in/load-out area where the gases mix with air circulating into the operating area exhaust system HEPA filter plenum and out of the CSB stack. These cask gases may contain suspended radioactive particulate that have leaked from the MCO into the cask. The cask may also contain residual particulate from the K Basin loading and cold vacuum drying unit operations. Radiation monitoring of the load-in/load-out area is provided by continuous air monitors (CAMs). Venting continues until cask pressure decreases to atmospheric pressure. During venting operations, a sample syringe may be used to collect a sample. A subsequent laboratory analysis measures for hydrogen concentration, water moisture, and other gaseous chemicals.

During venting operations, the servicing system automatically injects inert gas into the gas flow to dilute the hydrogen concentration to safe levels below the flammability limit. This dilution prevents a flammable mixture from reaching the operating area exhaust system. The servicing system proportionally controls the inert gas flow based on the pressure of the cask gases being vented to ensure efficient mixing and dilution. After reducing the cask pressure to atmospheric pressure, the cask can be pressure-purged with inert gas by pressurizing the cask with inert gas and venting through the cask servicing system to atmospheric pressure. This pressurizing and venting process may be repeated several times until the hydrogen concentrations within the cask are below the flammability level of hydrogen in air. After the pressure-purging and venting cycles, the hydrogen concentration within the cask will be low enough for safe removal of the MCO from the cask and placement in an overpack tube for monitored storage. If the preferred option is storing an MCO in the cask for a short time period (see Section A2.5.3) or shipping the MCO, the MCO cask may be repressurized with inert gas to ensure maintaining an inert atmosphere and preventing air in-leakage.

Table A2-2 lists the operating conditions for which the cask servicing operations equipment has been designed.
### Table A2-2. Cask Servicing System Design and Operating Conditions.

<table>
<thead>
<tr>
<th>Component</th>
<th>System design conditions</th>
<th>System operating conditions</th>
<th>Design operating environment</th>
<th>Operating environment</th>
<th>Material</th>
<th>Corrosion allowance</th>
</tr>
</thead>
<tbody>
<tr>
<td>Secondary HEPA filter (FH-2)</td>
<td>150 lb/in² gauge, 400 °F</td>
<td>80 lb/in² gauge, 150 °F</td>
<td>5% to 95% RH 55 °F to 95 °F</td>
<td>60 °F to 85 °F</td>
<td>304L stainless steel ASTM A176</td>
<td>None</td>
</tr>
<tr>
<td>1-in. supply connection</td>
<td>2,500 lb/in² gauge, -20 °F to 200 °F</td>
<td>2,000 lb/in² gauge, -20 °F to 115 °F</td>
<td>-27 °F to 115 °F</td>
<td>-20 °F to 115 °F</td>
<td>Carbon steel ASTM A53® S160</td>
<td>0.065 in.</td>
</tr>
<tr>
<td>HE-2 in.-HE-040-M, HE-2 in.-HE-046-M</td>
<td>2,500 lb/in² gauge, -20 °F to 200 °F</td>
<td>2,000 lb/in² gauge, 55 °F to 95 °F</td>
<td>5% to 95% RH 55 °F to 95 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel ASTM A53® S160</td>
<td>0.065 in.</td>
</tr>
<tr>
<td>HE-1-HE-036-L, HE-1-HE-062-L, MCO-12@MSS-041-L</td>
<td>150 lb/in² gauge, -20 °F to 400 °F</td>
<td>5 to 80 lb/in² gauge, -20 °F to 200 °F</td>
<td>5% to 95% RH 55 °F to 95 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel ASTM A53® S80</td>
<td>0.065 in.</td>
</tr>
<tr>
<td>MCO-12-MSS-039-SS-JS</td>
<td>150 lb/in² gauge, -27 °F to 400 °F</td>
<td>5 to 80 lb/in² gauge, -20 °F to 200 °F (normal) 400 °F (peak)</td>
<td>5% to 95% RH 55 °F to 95 °F</td>
<td>60 °F to 85 °F</td>
<td>304L stainless steel ASTM A312® S40</td>
<td>None</td>
</tr>
</tbody>
</table>

This table is for information only and does not provide operational controls.


HEPA = high-efficiency particulate air (filter).
RH = relative humidity.
Safety Considerations. For the transportation cask servicing system, the flex connector, HEPA filter, piping between the filter and the cask, and PSV-102 are classified as safety significant. The remaining components are general service (Figure A2-24).

A2.5.1.3 Multi-Canister Overpack Handling Machine Operations. The MHM (Figure A2-26) is designed for safe handling of an MCO in the operations area of the CSB. The MHM removes an MCO from the cask in the cask receiving pit and carries the MCO to a standard storage tube for interim storage, to the sampling/weld station for sampling/weld operations, or to an overpack storage tube. The first step in placing an MCO into a storage tube is the removal of the storage tube cover. The MHM, containing the MCO retrieved from the cask receiving pit, moves on its gantry crane to a position above the storage tube. The seismic restraints are set. The MHM retractable shield skirt is lowered to meet the operating deck to form shielding. The MHM turret is rotated until the storage tube plug cavity is above the storage tube plug, and the tube plug is raised into the tube plug cavity. Once the tube plug is fully raised, the MHM turret rotates until the MCO cavity is aligned with the storage tube, the turret locking pins are set, and the MCO is lowered into the storage tube. When the MCO is on the bottom of the storage tube or on top of an intermediate impact absorber, the MHM grapple disengages and withdraws, the MHM barrel rotates to place the tube plug cavity over the storage tube, the tube plug is lowered into place, the turret is rotated to the camera position, the seismic restraints are released, and the MHM moves away. The storage tube cover is replaced, and staging operations are complete. In addition, the MHM retrieves intermediate impact absorbers from the MHM maintenance pit and lowers them into the standard storage tubes before retrieving MCOs from the cask receiving pit.

Major Components. The following major components are used during MCO operations:

- MHM bridge and trolley
- MHM cask and turret
- MHM retractable nose unit and shield skirt
- MHM ventilation and filtration system
- MHM power and compressed air supply.

Equipment Description and Operational Considerations. A bottom impact absorber is placed in each standard and overpack storage tube before placement of the MCO. Also, before the MHM can insert an MCO into a tube or remove an MCO from a storage tube, the storage tube cover is removed and an interface guide ring funnel (Figure A2-27), constructed of ASTM A36 carbon steel, is manually installed on the top flange of the storage tube. An HPT surveys the storage tube cover while it is being removed and surveys the top of the tube plug assembly. The storage tube cover will be decontaminated if excessive contamination is found. Table A2-3 lists the operating conditions for which the standard storage tube has been designed.
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<table>
<thead>
<tr>
<th>Table A2.3</th>
<th>Standard Storage Tube and Multi-Canister Overpack Handling Machine Design and Operating Conditions</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Component</strong></td>
<td><strong>System design conditions</strong></td>
</tr>
<tr>
<td>Storage tube</td>
<td>0 lb/in² gauge, 220°F</td>
</tr>
<tr>
<td>Tube plug body</td>
<td>0 lb/in² gauge, 204°F</td>
</tr>
<tr>
<td>Tube plug seal</td>
<td>-65°F to 300°F</td>
</tr>
<tr>
<td>Tube plug piping</td>
<td>300°F, 150 lb/in², 0°F</td>
</tr>
<tr>
<td>Intermediate impact absorber</td>
<td>320°F, 0°F</td>
</tr>
<tr>
<td>Lower impact absorber</td>
<td>330°F, 10°F</td>
</tr>
<tr>
<td>Interface guide ring</td>
<td>60°F to 104°F</td>
</tr>
</tbody>
</table>

This table is for information only and does not provide operational controls.

**Abbreviations:**
- API = American Petroleum Institute
- ANSI = American National Standards Institute
- AISC = American Institute of Steel Construction
- ASTM = American Society for Testing and Materials
- MRM = multi-canister overpack handling machine

*NA* = not applicable.
The MHM is a fully shielded machine designed to safely transport an MCO within the CSB facility service and operating areas, including MCO retrieval from the cask receiving pit, the sampling/weld station pit, and the storage tubes. The MHM also removes and replaces storage tube plugs, the service station shield hatch plate, the sampling station shield plug, and impact absorbers.

The MHM is shown in Figure A2-26. The MHM comprises a bridge and trolley system and a shielded cask and turret system (Figure A2-28). The MHM weighs approximately 990,000 lb and is 17 ft high from the operating deck to the top of the trolley rails. The MHM turret and MCO hoist enclosure extend above this height. The MHM gantry bridge structure spans 126 ft, 6 in. from rail center to rail center. The wheel base is 23.5 ft center-to-center of the outside wheels, with two sets of 36-in.-diameter wheels on 60-in. centers. The MHM has a 2-in. nominal deck clearance below the nose of the cask with the shield skirt lifted above the deck, and a 9 ft, 0 in. minimum clearance between the deck surface and the bottom of the MHM gantry bridge box beams.

The SNF Project CSB uses the MHM to transport the MCOs. The MHM is designed to ride on steel rails that run north-south and are located at the 709 ft, 0 in. floor elevation. Anchor bolts are installed in the deck at the east and west sides of the operating floor for the installation of MHM rails. The MHM rails are installed in trenches that run north-south and cut across the receiving crane rails. The embedded MHM rails are provided with lateral restraints in addition to the anchor bolts. The lateral restraints and anchor bolts are designed to provide restraint of the MHM rails during the design basis earthquake. The receiving crane rails are installed in trenches that run east-west in the northernmost portion of the operating floor. The reinforced concrete construction of the northernmost portion of the operating floor includes the load-in/load-out area. Special rail crossover components called “frogs” are provided where the MHM rails and receiving crane rails intersect.

MHM operation is governed by a system of electrical interlocks. These interlocks (Table A2-4) provide permissives to the operator that allow the operator to perform MHM operations (e.g., actuate drives, motors) based on signals from a large number of sensors and switches. As can be seen in Table A2-4, many of the interlocks prevent or preclude MHM operational errors that may result in radiological consequences or facility damage. These interlocks are typically provided with two separate interlock protection channels (X- and Y-channel), thus providing single-failure-proof protection. Chapter A4.0 and the accident analysis sections in Chapter A3.0 identify MHM interlocks that perform safety-significant defense-in-depth functions. The operator station has indicator lamps, displays, and alarms that inform personnel about the state of the MHM. The operator will input commands to actuate drives via push buttons, switches, and joysticks.
### Table A2-4. Multi-Canister Overpack Handling Machine Interlocks. (10 sheets)

<table>
<thead>
<tr>
<th>Interlock</th>
<th>Interlock function</th>
</tr>
</thead>
<tbody>
<tr>
<td>P1</td>
<td>When the MHM handwind is being used, interlock P1 inhibits 480 V drive power and operation of the following:</td>
</tr>
<tr>
<td></td>
<td>- Bridge travel motor</td>
</tr>
<tr>
<td></td>
<td>- Crane bridge seismic clamps</td>
</tr>
<tr>
<td></td>
<td>- Trolley travel motor</td>
</tr>
<tr>
<td></td>
<td>- Trolley restraint pin and jack motors</td>
</tr>
<tr>
<td></td>
<td>- Shield skirt jack motor</td>
</tr>
<tr>
<td></td>
<td>- Turret rotate motor</td>
</tr>
<tr>
<td></td>
<td>- Turret locking pin</td>
</tr>
<tr>
<td></td>
<td>- Base locking pin</td>
</tr>
<tr>
<td></td>
<td>- Tube plug hoist and grapple</td>
</tr>
<tr>
<td></td>
<td>- MCO hoist.</td>
</tr>
<tr>
<td>P2</td>
<td>X-Channel: Unless the MHM is in the MCO mode or impact absorber exchange mode, the tube plug hoist is fully raised and the tube plug grapple jaws are fully open and the plug cask is empty AND unless the MHM is in the tube plug exchange mode and the tube plug hoist is fully raised and the tube plug grapple jaws are fully open and the plug cask is empty and the MCO grapple is in contact with a load and the MCO hoist weight is above the minimum grapple + tube plug weight limit OR the MHM is in the tube plug exchange mode and the tube plug hoist is fully raised and the tube plug grapple jaws are fully closed and locked and the plug cask is occupied and the MCO grapple is not in contact with a load and the MCO hoist weight is below the maximum grapple-only weight limit, the X-channel inhibits the following:</td>
</tr>
<tr>
<td></td>
<td>- Crane bridge travel</td>
</tr>
<tr>
<td></td>
<td>- Crane trolley travel</td>
</tr>
<tr>
<td></td>
<td>- Shield skirt jack raising.</td>
</tr>
<tr>
<td></td>
<td>Y-Channel: Unless the MHM is in the MCO mode or impact absorber exchange mode, the tube plug hoist is fully raised and the tube plug grapple jaws are fully open and the plug cask is empty AND unless the MHM is in the tube plug exchange mode and the tube plug hoist is fully raised and the tube plug grapple jaws are fully open and the plug cask is empty and the MCO grapple is in contact with a load and the MCO grapple load is not an MCO, OR the MHM is in the tube plug exchange mode and the tube plug hoist is fully raised and the tube plug grapple jaws are fully closed and locked and the plug cask is occupied and the MCO grapple is not in contact with a load and the MCO grapple jaws are not closed, the Y-channel inhibits the following:</td>
</tr>
<tr>
<td></td>
<td>- Crane bridge travel</td>
</tr>
<tr>
<td></td>
<td>- Crane trolley travel</td>
</tr>
<tr>
<td></td>
<td>- Shield skirt jack raising.</td>
</tr>
</tbody>
</table>
Table A2-4. Multi-Canister Overpack Handling Machine Interlocks. (10 sheets)

<table>
<thead>
<tr>
<th>Interlock</th>
<th>Interlock function</th>
</tr>
</thead>
<tbody>
<tr>
<td>P3</td>
<td>Unless the shield skirt jack is fully raised, interlock P3 (channels X and Y) inhibits the following:</td>
</tr>
<tr>
<td></td>
<td>- Crane bridge travel operating and crane bridge seismic clamps releasing and crane trolley travel operating and trolley restraint pins retracting and trolley restraint jacks retracting.</td>
</tr>
<tr>
<td>P4</td>
<td>Unless all the crane bridge seismic clamps are fully released, interlock P4 inhibits the following:</td>
</tr>
<tr>
<td></td>
<td>- Crane bridge travel.</td>
</tr>
<tr>
<td>P5</td>
<td>X-Channel: Inhibits crane bridge travel northward unless the north travel switch is not operated.</td>
</tr>
<tr>
<td></td>
<td>Y-Channel: Inhibits crane bridge travel unless the redundant overlap zone switch is not operated.</td>
</tr>
<tr>
<td>P6</td>
<td>Unless in MCO mode or impact absorber exchange mode and the MCO grapple is at the upperdatum</td>
</tr>
<tr>
<td></td>
<td>OR</td>
</tr>
<tr>
<td></td>
<td>if in the tube plug exchange mode and the MCO tube plug is at the tube plug raise limit, interlock P6 (channels X and Y) inhibits the following:</td>
</tr>
<tr>
<td></td>
<td>- Turret rotating and turret locking pin disengagement and base locking pin disengagement.</td>
</tr>
<tr>
<td>P7</td>
<td>Unless the tube plug hoist is fully raised, interlock P7 (channels X and Y) inhibits the following:</td>
</tr>
<tr>
<td></td>
<td>- Turret rotating</td>
</tr>
<tr>
<td></td>
<td>- Turret locking pin retracting</td>
</tr>
<tr>
<td></td>
<td>- Base locking pin disengaging.</td>
</tr>
<tr>
<td>P8</td>
<td>Unless the turret is aligned at the navigation camera position, interlock P8 (channels X and Y) inhibits the following:</td>
</tr>
<tr>
<td></td>
<td>- Shield skirt jack raising.</td>
</tr>
<tr>
<td>P9</td>
<td>Unless the base and turret locking pins are fully engaged, interlock P9 (channels X and Y) inhibits the following:</td>
</tr>
<tr>
<td></td>
<td>- Crane bridge travel</td>
</tr>
<tr>
<td></td>
<td>- Crane trolley travel</td>
</tr>
<tr>
<td></td>
<td>- Shield skirt jack raising</td>
</tr>
<tr>
<td></td>
<td>- Tube plug hoist operating</td>
</tr>
<tr>
<td></td>
<td>- MCO hoist operating.</td>
</tr>
<tr>
<td>P10</td>
<td>X-Channel: Inhibits crane bridge travel northward unless the north collision detect is not deflected.</td>
</tr>
<tr>
<td></td>
<td>Y-Channel: Inhibits crane bridge travel and crane trolley travel unless all collision detect bumpers are not deflected or the anti-collision detect override switch is held in the override position.</td>
</tr>
</tbody>
</table>
### Table A2-4. Multi-Canister Overpack Handling Machine Interlocks. (10 sheets)

<table>
<thead>
<tr>
<th>Interlock</th>
<th>Interlock function</th>
</tr>
</thead>
</table>
| P11       | If the crane rail frog is not aligned with the MHM crane rail, interlock P11 inhibits the following:  
- MHM crane bridge travel into overlap zone. |
| P12       | Unless the crane bridge is stationary, interlock P12 inhibits the following:  
- Application of crane bridge seismic clamps unless integral motor brakes are applied. |
| P13       | Unless the trolley seismic restraints are fully released, interlock P13 inhibits the following:  
- Crane trolley travel. |
| P14       | X channel: Inhibits crane trolley travel eastward beyond its east limit  
AND  
Inhibits crane trolley travel westward beyond its west limit.  
Mechanical interlock: Bumpers are fitted at each end of the bridge for end-of-travel restraint. |
| P15       | Unless the crane trolley is stationary, interlock P15 inhibits the following:  
- Trolley restraint pin insertion  
- Trolley restraint jack extending  
- Application of trolley travel integral motor brakes and thus application of trolley seismic restraints. |
| P16a      | Inhibits trolley restraint pin inserting  
UNLESS  
the trolley restraint pin torque limiter is not operated  
OR  
the trolley restraint pin is fully retracted  
AND  
the trolley restraint pin torque limiter reset timer has not timed out. |
| P16b      | Inhibit trolley restraint pin inserting  
UNLESS  
the trolley restraint pin is not fully inserted. |
| P16c      | Inhibit trolley restraint pin inserting  
UNLESS  
the trolley restraint jack is not extending. |
| P17a      | Inhibits trolley restraint jack extending if the trolley restraint pin is not at the intermediate position. |
| P17b      | Inhibits trolley restraint jack extending if its torque limiter is operated (except for resetting at the fully retracted position). |
| P18       | Inhibits trolley restraint jack retracting unless the trolley restraint pin is fully retracted. |
| P19       | Inhibits the shield skirt jack operating if its torque limiter is tripped. |
| P20       | Inhibits the shield skirt jack lowering below its lower limit. |
### Table A2-4. Multi-Canister Overpack Handling Machine Interlocks. (10 sheets)

<table>
<thead>
<tr>
<th>Interlock</th>
<th>Interlock function*</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>P21</strong></td>
<td>Unless the bridge crane seismic clamps are fully applied and trolley restraint pins fully inserted, interlock P21 (channels X and Y) inhibits the following:</td>
</tr>
<tr>
<td></td>
<td>- Shield skirt jack lowering</td>
</tr>
<tr>
<td></td>
<td>- Turret rotating</td>
</tr>
<tr>
<td></td>
<td>- Turret locking pin disengaging</td>
</tr>
<tr>
<td></td>
<td>- Base locking pin disengaging</td>
</tr>
<tr>
<td></td>
<td>- Tube plug hoist operating</td>
</tr>
<tr>
<td></td>
<td>- MCO hoist operating</td>
</tr>
<tr>
<td><strong>P22</strong></td>
<td>Inhibits the shield skirt jack raising above its upper limit.</td>
</tr>
<tr>
<td><strong>P23</strong></td>
<td>Deleted*</td>
</tr>
<tr>
<td><strong>P24</strong></td>
<td>Channel X: Inhibits turret rotation beyond its normal rotation limits. Mechanical interlock: Substantial end-of-travel bumpers are fitted in place.</td>
</tr>
<tr>
<td><strong>P25</strong></td>
<td>Inhibits turret rotation unless the turret and base locking pins are both fully disengaged.</td>
</tr>
<tr>
<td><strong>P26</strong></td>
<td>X-Channel: Unless the shield skirt jack is fully lowered and the shield skirt is seated, the X-channel inhibits the following:</td>
</tr>
<tr>
<td></td>
<td>- Turret rotating</td>
</tr>
<tr>
<td></td>
<td>- Turret locking pin disengaging</td>
</tr>
<tr>
<td></td>
<td>- Base locking pin disengaging</td>
</tr>
<tr>
<td></td>
<td>- Tube plug hoist operating</td>
</tr>
<tr>
<td></td>
<td>- MCO hoist operating.</td>
</tr>
<tr>
<td><strong>P27</strong></td>
<td>Deleted*</td>
</tr>
<tr>
<td><strong>P28</strong></td>
<td>Deleted*</td>
</tr>
<tr>
<td><strong>P29</strong></td>
<td>Releases the turret rotate brake if the turret rotate motor is operating or if the turret locking pin is engaging.</td>
</tr>
<tr>
<td><strong>P30</strong></td>
<td>Inhibits the turret locking pin disengaging beyond its fully disengaged limit.</td>
</tr>
<tr>
<td><strong>P31</strong></td>
<td>Unless the base locking pin is fully disengaged, interlock P31 inhibits the following:</td>
</tr>
<tr>
<td></td>
<td>- Turret locking pin disengaging.</td>
</tr>
<tr>
<td><strong>P32</strong></td>
<td>Inhibits the turret locking pin engaging beyond its fully engaged limit.</td>
</tr>
</tbody>
</table>
Table A2-4. Multi-Canister Overpack Handling Machine Interlocks. (10 sheets)

<table>
<thead>
<tr>
<th>Interlock</th>
<th>Interlock function</th>
</tr>
</thead>
</table>
| P33       | Unless the turret is stationary, interlock P33 inhibits the following:  
  - Turret locking pin engaging  
  - Base locking pin engaging. |
| P34       | Unless the turret is aligned at a recognized working position, interlock P34 (channels X and Y) inhibits the following:  
  - Turret locking pin engaging  
  - Base locking pin engaging. |
| P35       | Inhibits the turret locking pin engaging with excessive force due to pin fouling (motor stall, overload relay). |
| P36       | Inhibits the base locking pin disengaging beyond its fully disengaged limit. |
| P37       | Inhibits the base locking pin engaging beyond its fully engaged limit. |
| P38       | Inhibits the base locking pin engaging unless the turret locking pin is fully engaged. |
| P39       | Inhibits the base locking pin engaging with excessive force due to pin fouling (motor stall, overload relay). |
| P40       | Unless the tube plug hoist is not at a lower limit and the tube plug grapple is not supported, interlock P40 inhibits the following:  
  - Tube plug hoist lowering below its lower limit. |
| P41       | Unless the turret is at the tube plug hoist position, interlock P41 inhibits the following:  
  - Tube plug hoist lowering  
  - Tube plug grapple jaws opening. |
| P42       | Inhibits the tube plug hoist raising above its upper limit. |
| P43       | Inhibits the tube plug hoist raising with excess weight or excess torque (channel X — motor overload relay; mechanical interlock — torque limiter). |
| P44       | Unless the tube plug grapple jaws are fully open or fully closed, interlock P44 (channels X and Y) inhibits the following:  
  - Tube plug hoist raising. |
| P45       | Unless the tube plug grapple jaws are fully open or fully closed and locked, interlock P45 (channels X and Y) inhibits the following:  
  - Tube plug hoist raising above the seating zone. |
| P46       | Deleted |
| P47       | Unless the tube plug is supported and at a correct set down height, interlock P47 (channels X and Y) inhibits the following:  
  - Tube plug grapple jaws opening. |
Table A2-4. Multi-Canister Overpack Handling Machine Interlocks. (10 sheets)

<table>
<thead>
<tr>
<th>Interlock</th>
<th>Interlock function</th>
</tr>
</thead>
</table>
| P48       | Unless the tube plug grapple jaws are correctly seated on the load, interlock P48 inhibits the following:  
  - Tube plug grapple jaws closing. |
| P49       | Interlock P49 inhibits the MCO hoist operating unless the MCO hoist grapple is in an appropriate seating zone or the MCO grapple jaws are not open, AND the MCO grapple is not in contact with a load or the MCO grapple jaws are open or the MCO grapple jaws are closed. |
| P50       | Deleted* |
| P51       | X-Channel: Inhibits MCO hoist operating faster than 7 ft/min (variable speed drive “fast” setting).  
  Y-Channel: Inhibits MCO hoist operation unless excess-speed monitor is not tripped. |
| P52       | X-Channel: Inhibits MCO hoist lowering unless MCO hoist position is above lower limit and the MCO grapple is not supported.  
  Y-Channel: Inhibits MCO hoist position unless MCO hoist position is not at ultimate lower limit. |
| P53       | X-Channel: Unless the MCO hoist weight is above its minimum grapple-only weight (X) AND the MHM is in its impact absorber exchange mode, or in the tube plug exchange mode, or the MCO grapple jaws are not closed or the MCO hoist weight is above the minimum grapple + MCO weight limit, or the MCO grapple is within an appropriate MCO seating zone AND the MHM is in the MCO mode, or in the impact absorber exchange mode or the MCO grapple jaws are not closed, or the MCO hoist weight is above the minimum grapple + tube plug weight limit, or the MCO grapple is within an appropriate tube plug seating zone, the X-channel inhibits the following:  
  - MCO hoist lowering.  
  Y-Channel: Unless the MCO hoist weight is above its minimum grapple-only weight (Y) or the MCO hoist weight override key switch is held in the override position, the Y-channel inhibits the following:  
  - MCO hoist operating. |
| P54       | X-Channel: Inhibits the MCO hoist lowering unless the turret is at the MCO hoist position.  
  Y-Channel: Inhibits the MCO hoist operating unless the turret is at the MCO hoist position. |
| P55       | Unless (a) when lowering empty, the MCO grapple is not passing through the shield ring and (b) when lowering with a load, the load is not passing through the shield ring, interlock P55 inhibits the following:  
  - MCO hoist lowering faster than creep speed. |
Table A2-4. Multi-Canister Overpack Handling Machine Interlocks. (10 sheets)

<table>
<thead>
<tr>
<th>Interlock</th>
<th>Interlock function</th>
</tr>
</thead>
<tbody>
<tr>
<td>P56</td>
<td>X-Channel: Unless the tube plug hoist is fully raised and the tube plug grapple jaws are fully closed and the plug cask is occupied OR the crane bridge is within the exchange facility zone and the crane trolley is within the exchange facility zone, the X-channel inhibits the following: • MCO hoist lowering.</td>
</tr>
<tr>
<td></td>
<td>Y-Channel: Unless the tube plug hoist is fully raised and the tube plug grapple jaws are fully closed and the plug cask is occupied OR the crane bridge is within the exchange facility zone and the crane trolley is within the exchange facility zone, the Y-channel inhibits the following: • MCO hoist operating.</td>
</tr>
<tr>
<td>P57</td>
<td>X-Channel: Unless the MCO grapple is not at the upper datum and the MCO hoist is below its upper limit, the X-channel inhibits the following: • MCO hoist raising.</td>
</tr>
<tr>
<td></td>
<td>Y-Channel: Unless the MCO hoist is not at its ultimate upper limit, the Y-channel inhibits the following: • MCO hoist operating.</td>
</tr>
<tr>
<td>P58</td>
<td>X-Channel: Unless the MCO hoist position is below the tube plug raise limit or the MHM is in the MCO mode or in the impact absorber exchange mode, the X-channel inhibits the following: • MCO hoist raising.</td>
</tr>
<tr>
<td></td>
<td>Y-Channel: Unless the MCO hoist position is below the tube plug ultimate raise limit or the MHM is in the MCO mode or in the impact absorber exchange mode or the MCO grapple is not in contact with a load, the Y-channel inhibits the following: • MCO hoist operating.</td>
</tr>
<tr>
<td>P59</td>
<td>Deleted*</td>
</tr>
<tr>
<td>P60</td>
<td>Deleted*</td>
</tr>
<tr>
<td>P61</td>
<td>X-Channel: Unless the MCO grapple is in appropriate seating zone or the MCO grapple jaws are locked, the X-channel inhibits the following: • MCO hoist raising</td>
</tr>
<tr>
<td></td>
<td>Y-Channel: Unless the MCO grapple is in appropriate seating zone or the MCO grapple jaws are locked, the Y-channel inhibits the following: • MCO hoist operating.</td>
</tr>
</tbody>
</table>
Table A2-4. Multi-Canister Overpack Handling Machine Interlocks. (10 sheets)

<table>
<thead>
<tr>
<th>Interlock</th>
<th>Interlock function</th>
</tr>
</thead>
</table>
| P62⁺ | X-Channel: Unless the MCO grapple jaws are closed or the MCO hoist weight is below the maximum grapple-only X-weight, the X-channel inhibits the following:  
  - MCO hoist raising. |
| | Y-Channel: Unless the MCO grapple jaws are closed or the MCO hoist weight is below the maximum grapple-only Y-weight or the MCO hoist weight override key switch is held in the override position, the Y-channel inhibits the following:  
  - MCO hoist operating. |
| P63⁺ | X-Channel: Unless the MCO hoist weight is below its maximum grapple + MCO weight limit (and below MCO hoist rated load) AND the MHM is in its MCO mode, or in the tube plug exchange mode, or the MCO hoist weight is below the maximum grapple + impact absorber weight limit AND the MHM is in the MCO mode, or in the impact absorber exchange mode, or the MCO hoist weight is below the maximum grapple + tube plug weight limit, or the X-channel inhibits the following:  
  - MCO hoist raising. |
| | Y-Channel: Unless the MCO hoist weight is below its rated load or the MCO hoist weight override key switch is held in the override position, the Y-channel inhibits the following:  
  - MCO hoist operating. |
| P64 | Unless the MHM is in the impact absorber exchange mode or in the tube plug exchange mode or the MCO grapple jaws are not closed or the MCO grapple load is an MCO, interlock P64 inhibits the following:  
  - MCO hoist raising. |
| P65⁺ | X-Channel: Unless the MHM is in the MCO mode or the MCO grapple jaws are not closed or the MCO grapple load is not an MCO, the X-channel inhibits the following:  
  - MCO hoist raising. |
| | Y-Channel: Unless the MHM is in the MCO mode or the MCO grapple jaws are not closed or the MCO grapple load is not an MCO, the Y-channel inhibits the following:  
  - MCO hoist operating. |
### Table A2-4. Multi-Canister Overpack Handling Machine Interlocks. (10 sheets)

<table>
<thead>
<tr>
<th>Interlock</th>
<th>Interlock Function</th>
</tr>
</thead>
<tbody>
<tr>
<td>P66&lt;sup&gt;3&lt;/sup&gt;</td>
<td>X-Channel: Unless the MCO grapple is supported and the jaws are unlocked or the MCO grapple jaws are not closed and the MCO grapple is within an appropriate seating zone, the X-channel inhibits the following:</td>
</tr>
<tr>
<td></td>
<td>· MCO grapple jaws opening.</td>
</tr>
<tr>
<td>P67</td>
<td>Y-Channel: Unless the MCO grapple is supported and the jaws are unlocked AND the MCO grapple is within an appropriate seating zone, the Y-channel inhibits the following:</td>
</tr>
<tr>
<td></td>
<td>· MCO grapple jaws opening.</td>
</tr>
<tr>
<td>P68</td>
<td>Unless the MCO grapple is seated on the load or the MCO grapple is not in contact with a load, interlock P68 inhibits the following:</td>
</tr>
<tr>
<td></td>
<td>· MCO grapple jaws closing.</td>
</tr>
<tr>
<td>Deleted&lt;sup&gt;6&lt;/sup&gt;</td>
<td></td>
</tr>
<tr>
<td>Deleted&lt;sup&gt;6&lt;/sup&gt;</td>
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<td>Deleted&lt;sup&gt;6&lt;/sup&gt;</td>
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<td>Deleted&lt;sup&gt;6&lt;/sup&gt;</td>
<td></td>
</tr>
<tr>
<td>P80&lt;sup&gt;4&lt;/sup&gt;</td>
<td>Unless the MHM is located in the exchange facility and the plug cask is empty and the MCO cask is empty and the shield skirt is fully seated, interlock P80 (channels X and Y) inhibits the following:</td>
</tr>
<tr>
<td></td>
<td>· Change of operating modes.</td>
</tr>
<tr>
<td>P81</td>
<td>Inhibits crane bridge travel faster than slow speed unless it is not approaching an end of travel.</td>
</tr>
<tr>
<td>P82</td>
<td>Inhibits retracting the trolley restraint pin if it is at its fully retracted limit.</td>
</tr>
<tr>
<td>P83</td>
<td>Inhibits retracting the trolley restraint jack if it is at its fully retracted limit.</td>
</tr>
<tr>
<td>P84</td>
<td>Inhibits operation of the tube plug grapple jaws unless the tube plug grapple jaws are unlocked.</td>
</tr>
</tbody>
</table>
Table A2-4. Multi-Canister Overpack Handling Machine Interlocks. (10 sheets)

<table>
<thead>
<tr>
<th>Interlock</th>
<th>Interlock function</th>
</tr>
</thead>
</table>
| P85       | X-Channel: Unless the MCO grapple load is not an MCO or the crane bridge is not within the exchange facility zone or the crane trolley is not within the exchange facility zone, the X-channel inhibits the following:  
  - Shield skirt jack lowering  
  - MCO hoist lowering. |
| P86       | Y-Channel: Unless the MCO grapple load is not an MCO or the crane bridge is not within the exchange facility zone or the crane trolley is not within the exchange facility zone, the Y-channel inhibits the following:  
  - Shield skirt jack lowering  
  - MCO hoist operating. |
| P87       | Inhibits the MCO hoist raising if the reeling drums do not operate. |
| P88       | Inhibits the MCO hoist raising if the rope loads are not balanced. |
| P89       | Deleted* |
| P90       | Deleted* |
| P91       | Deleted* |
| P92       | Inhibits the MCO hoist if a rope is misreeving. |
| P93       | Inhibits cask extract fan if it is producing low flow after initial startup period. |
| P94       | Inhibits cask extract fan if its filter is becoming blocked. |
| P95       | Inhibits camera lights if cask extract fan is not running. |
| P96       | Inhibits operation of camera lights if the turret is not at the camera position or the turret is at the camera position and the time delay has expired. |

This table is for information only and does not provide operational controls.


The previous safety function of this interlock is no longer needed.

MCO = multi-canister overpack.

MHM = multi-canister overpack handling machine.
The MHM is designed to prevent MHM lateral displacement that would shear an MCO as it is being raised or lowered inside the throughport. A system of MHM drive control interlocks provides for safe operation of the MHM. Interlocks prevent inadvertent activation of both bridge drive motors and the trolley drive motor unless the interlock sensors report that the state of the MHM is appropriate to allow safe operation for the motors. This interlock-permissive system greatly reduces the chance of (or in some cases prevents) damage to an MCO caused by personnel error, NPH, or failed protection channels.

**Bridge and Trolley System.** The MHM bridge and trolley system comprises a double-girder bridge with a top-running trolley supported on four end trucks that have two wheels each. The crane bridge rides on north-south runway rails recessed in the operating deck. The bridge drive truck system on each south girder (two motors, 7.5 hp each) is geared down to 244:1. Associated with each of the bridge trucks (four total) is braking from the flux vector-controlled brakes and end-of-travel switches. The trolley travels east-west on rails mounted on top of the bridge. The shielded cask and turret system is rigidly supported on the top-running trolley. This arrangement allows the cask to be positioned over all of the MCO storage and handling locations throughout the operating area. The 15 hp trolley drive system also has flux vector-controlled braking and end-of-travel switches. The trolley drive motor and the bridge drive motors have variable frequency motor controllers that allow operation of the motors at variable speeds. These variable frequency motor controllers have redundant contactors (X- and Y- channels), thus providing single-failure-proof protection.

The crane rails and two rail frogs (Figure A2-29) provide the interface surface for the four trucks and for the bridge seismic rail clamps. The rail frogs (Figure A2-9) are designed to bridge the gap at the rail intersections of the receiving crane and MHM rails. Each frog makes up the missing section of rail for one rail system (and corresponding machine) while placing a rail arm with a wedge stop along the rail in the direction of the other rail system (and corresponding machine). The wheels of the other machine (MHM or receiving crane) are stopped by the mechanical stop if they attempt to enter the overlap zone with the rail frog not aligned toward them. The rail frog at the southwest corner of the receiving area is equipped with a striker to actuate a limit switch on the MHM, which will monitor the orientation of the rail frog as the MHM attempts to cross into the receiving area. If the rail frog orientation is incorrect, a limit switch will interrupt power to the MHM bridge and stop MHM movement.

Seismic clamps located in the bridge wheel trucks grip the MHM crane rail flanges when deenergized. The crane bridge and trolley restraints ensure that the bridge is held in place on the runway rails and the trolley stays in its location on the trolley rails on the bridge girders during a seismic event. Several interlocks depend on the seismic clamps being engaged with the rails before permitting various functions of the MHM to take place. The fail-safe bridge truck clamps have shoes that are applied by coil springs and hydraulic motors that are energized to disengage the clamps. The redundant seismic clamp position indication is interlocked with the MCO hoist limit switches and inhibits power to the clamp hydraulic pump if the hoist is not at its top position. Interlocks also prevent the trolley brakes from being applied when the trolley bridge is moving. A hydraulic flow orifice prevents a rapid depressurization of the hydraulic system on the bridge.
truck seismic restraint clamps, and thus clamp engagement, before the MHM has coasted to a stop on loss of electrical power.

Two seismic X-restraint pins (Figure A2-30) lock the trolley to the bridge rails, preventing east-west motion of the trolley and turret. When activated, trolley pins are inserted into pockets welded onto the bridge beam. Personnel visually verify that the trolley seismic restraint pins (Figure A2-30) are fully inserted before using the MCO hoist. The trolley south seismic pin can be seen by standing on the trolley ladder. The trolley north seismic pin can be seen using a mirror mounted near the MHM control cabinets on the operating platform. These locking pins have the shear strength to safely withstand seismic forces during a design basis earthquake. Restraints for the trolley in the north-south direction are passive restraints formed by the wheel flanges. Uplift restraint is provided by four fixtures that hook beneath the top flange of the girders. The seismic pin power is interlocked with the MCO hoist limit switches and inhibits power to the locking pin drive motor to disengage the pin if the hoist is not at its top position. These locking pin drive systems have position switches that permit power to the MCO hoist only on redundant indication of full insertion.

A collision-avoidance system is provided to stop the MHM when it collides with objects on the deck of the CSB that could interfere with the movement of the MHM. Interlock P10 stops trolley travel when the east-west bumpers are activated and stops bridge travel when the north-south bumpers are activated. The MHM collision-avoidance system uses an octagonal arrangement of pressure-sensitive bumpers with redundant sensors surrounding the retractable shield skirt at the bottom to detect by contact objects that are in the path of the MHM. The MHM collision-avoidance system actuates a signal to the MHM control system once the object is detected (bumped), and the P10 interlock prevents movement of the MHM toward the detected object. The P10 interlock inhibits crane bridge travel and prevents crane trolley travel collision with the sample station gantry crane, the service tent, or an object greater than 2 in. high standing on the operating deck.

The receiving crane and MHM anti-collision systems prevents the MHM and the receiving crane from colliding when they operate at the north end of the CSB operating area. When one crane or the other is in the overlap zone where a collision could occur, SPSSs (Figure A2-23, sheet 3) are tripped (de-energized) to restrict operation of the other crane so that it cannot travel into the same area. When the MHM enters the overlap zone, the MHM toggles two RALSs, (Figure A2-23, sheet 4) which removes power from the two SPSSs mounted on the receiving crane's power strip structural support. The SPSSs are spring loaded and rotate striker arms to a position that will toggle RALSs on the receiving crane. Actuation of the receiving cranes RALSs, if the receiving crane attempts to enter the overlap zone, removes power to the receiving crane bridge drive motors and the receiving crane is prevented from entering the overlap zone. The P5 MHM interlock removes power to the bridge drive motor of the MHM if it attempts to enter the receiving area when it is already occupied by the receiving crane. As noted earlier, the rail frog mechanical stop also prevents the MHM and receiving crane from colliding.
The MHM travel limit switches include features that prevent the MHM from traveling too far north or south and a safety system for preventing the MHM turret from rotating to the MCO position while at the MHM maintenance pit or sampling/weld station area. Limit switches mounted on the east MHM tie beam deck (Figure A2-23, sheet 2) are toggled by the MHM north slow and stop limit switch striker arm (Figure A2-23, sheet 4) if the MHM travels too far north. The striker arm first toggles the north slow limit switch, slowing the MHM speed, and then, if the MHM continues to travel north, toggles the north stop switch to remove power from the MHM bridge drive motors. Similarly, the south slow and stop switch striker arm slows and stops MHM travel if the MHM attempts to travel too far south. The MHM maintenance pit limit switch system consists of RALSs mounted on the east MHM tie beam deck (Figure A2-23, sheet 4) and ramps mounted on supports (Figure A2-23, sheet 4). As the MHM travels north, the upper RALSs are toggled by the exchange facility ramps that inhibit turret rotation when the trolley travels over the MHM maintenance pit. This system reduces the risk of dropping an MCO into the MHM maintenance pit. A similar single RALS and ramp system at the south end of the operation area warns the MHM operator if the sampling/weld station covers are not in place before operating the MHM in the area.

**Cask and Turret System.** The integrated cask and turret system allows the MHM to:

1. locate the cask receiving station, a storage tube, or a sampling/weld station;
2. remove the shield hatch plate, the tube plug, or the sampling/weld station center shield plate;
3. retrieve and transport an MCO from the transport cask; 
4. place an MCO in, or remove and transport an MCO from, a tube or sampling/weld station; 
5. replace the tube plug, shield hatch plate, or center shield plate; and 
6. place an intermediate impact absorber in the tube (note Figures A2-31 and A2-32). The cask and turret system comprises a stack of interconnected components that is rigidly suspended from the trolley. The trolley supports a rotating upper turntable, which supports the upper turret and hoist assembly. A stationary lower turntable and retractable shield skirt are fixed below the rotating turret by the base torsion link arm from below the trolley. The two rotating points use large concentric slewing bearings and a large ring gear that are mounted concentrically. The upper turret and the turntable can spin freely around the vertical centerline established by the two bearings. Rotation of both components is controlled with operational interlocks and seismic restraints, especially during MCO raising and lowering operations. The turret rotate drive motor and the MCO hoist motor have variable frequency motor controllers that allow operation of the motors at variable speeds.

The upper turret has three main cavities:

- An MCO cavity inside the main cask body, with an MCO hoist and grapple that is used for raising and lowering an MCO or an impact absorber and, using a plug adapter grapple, replacing a tube plug
- A plug cavity, with a hoist and grapple system to accommodate routine plug handling, to facilitate replacement of damaged plugs, and to raise and lower the shield hatch plate and center shield plate
A radiation-hardened television camera with a zoom lens for viewing down a storage tube via the television monitoring system, and sufficient lighting for accurately positioning the MHM on the appropriate targets over the cask receiving station, above a storage tube, or at a sampling/weld station, and for viewing down a storage tube.

These three cavities each have a vertical centerline spaced equidistant from the vertical centerline of turret rotation in such a way that each may be indexed to precisely coincide with the single throughport in the lower nonrotating turntable. The main cask body represents the largest mass in the upper turret. Placement of the cask body and plug cavity housing means that the upper turret has a heavy mass center of gravity eccentrically located to the vertical line of rotation. Figure A2-33 provides a cross-sectional view of the MHM turret showing the MCO and tube plug cavities.

The upper turret can be rotated into the following three index positions: (1) MCO cavity, (2) tube plug cavity, and (3) television camera. The upper turntable and drive motor are used to rotate the MHM turret to the desired orientation. The lower turntable has a single throughport for passage of an MCO, tube plug, or impact absorber. The throughport is on the same radius as the three cavities eccentrically located with respect to the rotation centerline of the two slewing bearings. This arrangement allows the turret to rotate with respect to the base such that one of three turret compartments may be aligned with the throughport. The lower ends of the MCO and tube plug cavities, which are “open” when MCOs or plugs are being transferred into or out of the MHM, are automatically closed whenever the upper turret is rotated away from the “MCO cavity” or “tube plug cavity” index positions.

The turret-locking pin (Figure A2-34) is a powered, 5-in.-diameter horizontal steel pin that fits into any of three pockets in the turret support turntable (Figure A2-35). When the pin is engaged in one of these three pockets, it serves to precisely align the turret with one of its three positions (MCO cavity, tube plug cavity, or television camera) and to prevent it from rotating by securing it to the trolley. Personnel visually verify that the turret-locking seismic pin, located on the same level as the MHM operating platform, is fully inserted before using the MCO hoist. This locking pin has the shear strength to safely withstand the seismic forces of a design basis earthquake. The single failure-proof interlock with the MCO hoist limit switches inhibits power to the locking pin drive motors to disengage the pin if the hoist is not at its top position. This locking pin drive system has a position switch that permits power to the MCO hoist only on satisfactory engagement of the locking pin.

A powered, base-locking pin (Figure A2-36) secures the rotating turret base to the stationary lower turntable at each turret index position. On the operating deck, an individual, while monitoring the MHM visually verifies that the base-locking pin seismic restraint is inserted before the MHM operator uses the MCO hoist. The base-locking pin has the shear strength to safely withstand seismic forces during a design basis earthquake. The locking pin drive has position indication switches that are interlocked with the MCO hoist and with the bridge, trolley,
and turret drives. The base-locking pin drive systems have switches that will permit power to the MCO hoist only on satisfactory engagement of the locking pins.

A television camera within the MHM camera cavity on the turret assembly is used to position the MHM and view inside the storage tubes. A single zoom-lens camera is mounted within the camera cavity along with six tungsten halogen lamps and a coarse fixed-lens backup camera. When the turret is rotated to the television camera, the main camera is aligned with the machine axis so that precise alignment between this axis and the desired MHM location may be achieved. The signal from either camera can be displayed on a 12-in. video monitor at the operator station console in the operator station on the MHM trolley. Using the image on the monitor, personnel can recognize when the MHM is properly aligned with the appropriate target. The camera also provides personnel with a means to read the unique alphanumeric identifier on the top face of each tube plug.

A tube plug hoist and grapple system (Figure A2-37) is used to raise and lower the storage tube plug. The MHM has the ability to lift the tube plug, or other similar plug (i.e., shield hatch plate or center shield plate), with the plug hoist, and to place it back into its fitted opening. During MCO retrieval or placement operations, the tube plug cavity temporarily houses the tube plug or other similar plug. Redundant optical sensors in the tube plug cavity detect the presence of a tube plug suspended by the tube plug hoist and close contacts which allow the turret to rotate to the MCO hoist position. The tube plug hoist is a screw jack driven by a single-speed AC motor that is vertically mounted. The interface is between the MHM plug grapple and the lifting pintle of the tube plug or other similar plug. MHM controls ensure that the grapple and hoist are able to hold the plug securely, without the ability to release it, except when the weight of the tube plug is supported. Mechanical roller cams engage the grapple jaws before raising the tube plug or until the tube plug is seated in the tube when lowering it. A mechanical torque limiter ensures that the force exerted by the jack cannot exceed design limits in any faulted condition. In the event of motor failure, a shoe brake prevents reverse drive and plug drop.

The MCO hoist and grapple system (Figure A2-38), located in the MCO cask within the turret (Figure A2-33), raises and lowers an MCO, a replacement storage tube plug, or an intermediate impact absorber. The MCO hoist and grapple system has a rated capacity of 12 tons and meets single failure criteria (HNF-S-0425). The upper limit of hoist travel is set by a limit switch (the high limit switch), which is tripped by the top of the MCO grapple. The elevation of the grapple is of utmost importance because it represents the only elevation at which the MCO can be considered safe from being accidentally impacted or trapped by inadvertent operation of the bridge, trolley, and/or turret drives. In addition to interrupting power to the hoist motor, tripping the high limit switch also permits the disengagement of the base and turret locking pins. This in turn permits rotation of the upper turret to the television camera, thereby automatically closing off the bottom opening of the MCO cavity. The high limit switch has a redundant backup, which is labeled the high-high limit switch. The MCO hoist has a resolver to indicate the instantaneous grapple height. An MCO centering guide mounted in the MCO cavity (Figure A2-33) helps to align the MCO when an MCO is lowered into a storage tube or a
The grapple is suspended by a redundant, two-part wire rope system. The MCO grapple (shown in Figure A2-38) is a self-centering, self-aligning, six-jaw grapple. The grapple is designed to engage and lift, and lower and release, the MCOs, impact absorbers, and tube plugs fitted with a lifting adaptor. The grapple uses its own weight to align the six jaws with the MCO lifting feature at all offset positions within the storage tube. An air-operated cylinder releases the jaws, allowing them to close under their own weight once the grapple is fully engaged and positioned. A positive mechanical lock on the locking plate ensures the jaws cannot be opened when the grapple is carrying a load. A manual, compressed air-actuated release mechanism for the jaws is available if normal-supply compressed air is unavailable. The manual connection for the air cylinder is located inside the hoist enclosure and accessible through hand holes at the intermediate platform.

Control and interface of the grapple is through duplicate limit switches. Two sets of limit switches are used to indicate the position of the probe. The first set of limit switches is set to operate once the control probe has been depressed approximately 0.375 in. (load contacted) to stop the hoist and initiate the jaws-open sequence to center the grapple. The second set of limit switches operates when the proximity probe is fully depressed by the locking plate, indicating the grapple is seated. This second set of switches enables discrimination between an MCO and an impact absorber or tube plug because each impact absorber and tube plug adaptor has a recessed center that can be sensed by the central probe in the MCO grapple.

Both the MCO and plug grapples are equipped with electrical and mechanical interlocks that prevent them from opening unless they are in the correct location and orientation. This is intended to eliminate the possibility of inadvertently releasing and dropping the MCO or the plug. The interlocks also prevent the hoists from raising or lowering unless the grapple position and status are as required for safe handling of the MCO and plug.

A festoon connects the CSB facility's service air system to the MHM. Valves in the grapple air supply system include a manual isolation valve to isolate the grapple air supply from the balance of the air supply system on the MHM. A pressure regulator controls the supply pressure to the grapple air supply system. A pressure switch provides pressure indication for the system and actuates and annunciates on low pressure. A check valve is provided to prevent back contamination of the air supply. A solenoid-actuated valve directs air to the grapple actuation cylinder to open the grapple jaws or relieves the pressure within the grapple actuation cylinder to close the grapple jaws. The solenoid-actuated valve is spring-fail to relieve pressure within the grapple actuation cylinder on loss of power. A backup air supply reservoir and valves are provided on top of the hoist enclosure (Figure A2-39).

The MCO hoist consists of dual rope systems, each capable of supporting the full load on a twin-groove drum that is connected to a primary and secondary planetary right angle gearbox, one on each side of the drum. The primary electric shoe brake is positioned between the motor...
and right angle gearbox and provides the main load-holding brake. A second brake provides dual
backup load-holding capability. Two handwinds and a height resolver and indicator are
connected via a bevel gearbox. The handwinds provide a means for raising or lowering the load
in the event of motor failure. The resolver provides a signal to the control system indicating
current grapple height. A dial indicator on the drum handwinds provides a rough mechanical
position indication of the grapple height.

The two hoist ropes travel from the drum up to twin head pulleys, down to the load block,
and back up to a balancing beam where they are fixed (Figure A2-38, sheet 2). Load cells on the
hoist, two per rope, indicate the load suspended from the grapple. Limit switches on the balance
beam detect whether its position is level, thereby maintaining balanced loads between the support
ropes. This arrangement detects any unbalanced loading or spooled length discrepancy between
the two ropes. A hydraulic cylinder connected to a strongback mounted on the floor of the FFTF
pit (Figure A2-9, sheet 3) will be connected to the MHM hoist periodically to provide verification
testing and calibration for the MHM MCO hoist weight system.

The hoist holding brakes support the MCO when hoisting is halted with the MCO
suspended. Each holding brake has a capacity to hold a load equal to 125% of the MCO hoist
rated load. The holding brakes are automatically applied on interruption of power to the hoist
motor by loss of brake power and spring-close shoes on the drum. For controlled lowering of the
MCO, the hoist is equipped with a flux-vector motor drive that provides normal braking to
control the descent at any speed within the specified speed range.

Retractable Nose Unit and Shield Skirt. The MHM shield skirt and the nose unit is a
suspended assembly that is raised when the MHM moves and is lowered when the MHM
performs MCO transfer operations. The shield skirt is a concrete slab resting on top of a steel
ring that suspends four rows of annular steel segmented floating blocks. Airflow to provide air
sweep of any contamination is drawn through the nose unit and shield skirt by the MHM extract
ventilation system. The MHM retractable shield skirt is lowered to rest on the deck and the nose
unit is lowered against an interface ring or pit cover that encircles the access opening. Interlocks
are provided to constrain the raising and lowering of the skirt and the nose to the proper
sequence. The shield skirt position is interlocked with bridge, trolley, and turret drives to
maintain shielding integrity and to prevent damage to the MCO, deck, or the MHM. The shield
skirt provides radiation shielding for the operating area during MCO raising, lowering, and
transferring operations.

Ventilation and Filtration System. To support ALARA goals, the MHM onboard filtration
(extraction) system (Figure A2-39) removes any unexpected incidental contamination that may be
present on an MCO. This system also helps to minimize contamination spread and cleanup costs
in a radiological release scenario. The MHM maintains an airflow up through the MCO turret
cavity (around the MCO) and out through a HEPA filter that exhausts to the operating area. An
alpha CAM, beta CAM, and a radioactive gas monitor can be mounted on the MHM trolley
exhaust or work area if radiation monitoring is necessary. A negative pressure within the MHM is
maintained relative to the operating area. The airflow provides cooling. Passive natural
convection cooling to the massive steel casting of the MCO cavity (a heat sink) will provide supplemental cooling of the MCO. The operating area has been designed to be maintained by the operating area exhaust system at a negative pressure relative to the outdoors and the support building. Therefore, any airborne contamination from the MHM will be exhausted through the operating area HEPA filters before reaching the environment.

**Electrical Power and Compressed Air Supply.** The MHM festoon system provides electrical power and a compressed air supply for the MHM. The part of the festoon system that accommodates bridge movement is suspended from a support rail mounted on the CSB wall columns. Electrical power for the MHM is provided from the northeast corner of the CSB through a 480 V main disconnect switch that is fed by motor control center (MCC) MC-32-209. Power is routed to the MHM bridge and trolley via the bridge festoon and the trolley festoon. The disconnect switch provides power to the MHM power loads.

Electrical service to the cask mechanisms is provided via the turret rotate festoon and a flexible connection between the trolley and rotating upper turret. One end of the flexible connection is suspended from the hoist platform and the other end is attached to the turret rotate festoon mounted on the trolley. The MHM power supply connection includes dual seismic contactors mounted to the northeast deck curb in the receiving area and actuated by two of three accelerometers in any one of three channels. The seismic power disconnect eliminates potential spurious operation resulting from seismic initiated spurious operation of the non-seismically qualified MHM electrical control system. The 480 V redundant power line contactors are located in a separate panel next to the accelerometer trip circuit panel. The power line contactors are fed by a signal from the accelerometer trip circuits based on a two-of-three trip signal polling circuit. If any contactor operates, the power circuit to the MHM will open. The trip setting for the seismic trip system is 0.19 g peak ground acceleration (Report QA8685) based on guidance in ASME NOG-1-1995 that the setpoint include the maximum horizontal and vertical spectra and 90% MHM weight participation at 7% damping. The estimated horizontal seismic spectral response acceleration values for CSB structures (5% damping) and cranes (7% damping) are approximately two times the 0.35 g peak ground acceleration horizontal value. The trip setting, therefore, is approximately 54% of the building and crane response acceleration values to the design basis earthquake. This setpoint trips power to the MHM to protect the MCO from damage during the postulated design basis earthquake. The panel containing the accelerometers is rigidly mounted and seismically qualified.

The CSB compressed-air supply line provides plant air to the MHM festoon cable line connection at approximately 100 lb/in² gauge. The compressed-air supply system provides air for gas cylinder actuation (opening) of the MCO grapple and also for cooling the MCO camera lights. The cooling airflow is sized to maintain ambient air temperature around the cameras while cooling the lights provided for the cameras. An onboard air receiver (Figure A2-39) is required to function as a backup compressed-air source to the manually actuated MCO grapple opening valve. This arrangement ensures that the manual opening function is always available. The air receiver is provided with an extra connection to facilitate pressurization using a portable air compressor in the event that plant air is unavailable.
System Operation. The MHM retrieves an MCO after the transportation cask containing
the MCO has been lowered into the cask receiving pit by the receiving crane. The sides of the
cask receiving pit are equipped with guides to align the cask so the MCO can be removed from
the cask by the MHM's MCO grapple and hoist. An impact absorber at the bottom of the pit
supports the cask at the correct elevation so the grapple and hoist will encounter the MCO at an
accessible height. The cask impact absorber also mitigates any damage to an MCO caused by a
cask drop.

The shield hatch and MCO guide assembly covers the cask receiving pit and provides
shielding. It consists of a shield hatch ring and a shield hatch plate. It serves as the physical
interface with the MHM retractable nose and shield skirt during MCO transfer operations
between the MHM and the transport cask. The center opening in the shield hatch ring is sized to
allow passage of an MCO. The diameter and design of the shield hatch plate are such that the
plate may be handled by the MHM in a way that is similar to handling the tube plug. A recessed
lift pintle is provided on the top center of the shield hatch plate for lifting by the MHM plug
grapple. The top of the pintle is flush with or below the surface of the operating deck. The lift
pintle is marked with a sighting target for interface with the MHM television camera to assist in
positioning the MHM. The shield hatch plate is stowed in the MHM plug chamber during MCO
raising operations. The MHM nose unit lowers to contact the shield hatch ring. The MHM
retractable shield skirt lowers to make up the shielding interface, contacting the concrete area
around the cask receiving pit, and protecting personnel from radiation exposure. The MHM
MCO grapple couples with the lifting flange at the top of the MCO, and the hoist raises the MCO
within the MCO cavity until the upper limit switch interlock is activated. The MHM then replaces
the center shield hatch plate and moves the MCO to the appropriate tube location or
sampling/weld station.

The tube plug cover plate must be separately removed from the deck embed before a
storage tube can be accessed. A small hoist attached to the tube vent and purge cart lifts the tube
plug cover plate and stows the plate temporarily in a safe, out-of-the-way location. If a storage
tube already is being used to store SNF, the material balance area custodian removes the tamper-
indicating device from the storage tube plug prior to access.

After removing the tube plug cover plate, personnel install a standard interface guide ring
funnel (Figure A2-27) using the tube vent and purge cart hoist. The standard interface guide ring
funnel (Figure A2-27) is a spacer placed between the storage tube and the MHM nose unit. The
standard interface guide ring funnel fits snugly into the top of the storage tube embed, resting on
the bellows assembly top flange. It fills the gap between the top of the bellows assembly and the
deck level. The standard interface guide ring funnel is lifted and handled using the tube vent and
purge cart hoist.

After preparing the storage tube for an MCO transfer operation, the operator drives the
MHM over the tube opening. Before moving the MHM, the MHM operator rotates the turret
and positions the television camera over the center of the throughport. The operator positions the
MHM over the tube and uses the television camera to accurately position the MHM's nose and
throughport directly over the tube centerline. The camera lighting system allows the operator to
clearly view the surfaces below the camera on a view screen located at the MHM console. The
camera view screen has a crosshair on it to aid the operator in centering the MHM over the tube.
The crosshair is lined up with a marking on the top center of the tube plug lift pintle. The
operator also views the tube plug number to verify that the proper tube is being accessed. The
MHM shield skirt is lowered to the operating deck so that adequate shielding is available before
tube plug removal.

With the MHM properly aligned over the tube, the operator sets the seismic restraints,
rotates the MHM turret, and aligns the tube plug cavity with the throughport. The tube plug
grapple is lowered and engages the tube plug pintle. Once properly engaged, the tube plug is
lifted into the MHM tube plug cavity, and the turret is rotated back to the television camera
position. The camera can be used to inspect the interior of the tube using a variable zoom lens
and supplemental lighting to view the MCO that is to be removed or to verify that an impact
absorber, bottom or intermediate, is in place before an MCO is lowered into the tube.

With the tube plug removed, the MHM performs one of three general tasks: place or
remove an MCO, place or remove an impact absorber, or replace the existing tube plug with a
new one. All of these tasks are performed using the MCO grapple and hoist. First the MHM
turret is rotated to the MCO cavity position. If an MCO, an impact absorber, or a new tube plug
is to be lowered into place, it will already be attached to the MCO grapple. The MHM operator
positions the console switches to automatically lower the MCO, impact absorber, or new plug to
the proper height. When the load has been relieved, the MCO grapple can be unlocked, opened,
and retrieved from the storage tube into the MCO cavity. If an impact absorber or MCO is to be
retrieved from the tube, the empty grapple is lowered into the tube. The hoist stops after a sensor
on the grapple has made solid contact with the MCO or impact absorber. The grapple opens and
self-centers itself in the tube as rollers on the outer jaw surface contact the inner wall of the tube.
The grapple is lowered until sensors indicate that it has fully engaged the object to be lifted. The
grapple is then closed, the jaws locked, and the MCO or impact absorber is removed from the
storage tube by the hoist. Unless the MHM is in the tube plug exchange mode, the tube plug in
the MHM tube plug chamber must be replaced in the tube before the shield skirt can be raised and
the MHM relocated.

If a tube plug is found to be damaged or faulty, the tube plug is exchanged with the aid of
the MHM. After the MCO centering guide is removed, a grapple adapter is installed on a
replacement tube plug at the plug exchange facility in the MHM maintenance pit and is picked up
by the MCO grapple and stowed in the lower part of the MCO cavity. The MHM moves to the
storage tube location, where it removes the faulty tube plug with the plug grapple, places the tube
plug in the MHM plug cavity, rotates to the MCO cavity, and lowers the replacement tube plug
into the storage tube using the MCO grapple.

Intermediate impact absorbers protect the first MCO in a standard storage tube from
damage caused by accidental dropping of the second MCO onto it. Intermediate impact
absorbers are required between the MCOs in the standard storage tubes. The intermediate impact
absorbers are designed for placement using the MCO grapple. The intermediate impact absorber
is retrieved from its storage place (an overpack tube or the exchange facility) and placed on top of
the bottom MCO in the storage tube before a second MCO is inserted into the tube. All
conditions that are required for placing an MCO into the storage tube must be met to place the
intermediate impact absorber. The impact absorbers are provided with a central recess that allows
the MCO grapple to recognize an impact absorber. The recess also provides a sighting target for
interface with the MHM television camera to assist in positioning the MHM for attaching the
grapple to the impact absorber.

The sampling/weld stations, located in pits at the south end of the CSB, are used to weld
the canister cover assemblies to the MCOs. The sampling/weld station located in the easternmost
pit (pit 7) will serve initially as a sampling station for monitoring of MCOs. The sampling/weld
station pit and trench shields are designed for docking with the MHM when the MCO transfers
are made between the MHM and the sampling/weld stations. The MHM moves over the
sampling/weld station pits and centers precisely over the shielding assembly for MCO transfer
operations. Precise alignment with the shielding assembly is accomplished by using the MHM
television camera and locating the target on the center shield plate pintle. The center shield plate
pintle is designed to be handled by the MHM plug grapple, and the center shield plate is stowed in
the plug cavity of the MHM during MCO lowering and raising operations. All equipment that
may be placed in the coverage area of the MHM, including the sampling/weld station gantry crane
and hoist, is less than 8 ft, 10 in. high to avoid interference with the MHM bridge.

MCO transfer at the sampling/weld stations is accomplished after lowering the MHM nose
unit onto the sampling/weld station shielding assembly and the shielding skirt against the deck
around the sampling/weld station pits. The MCO grapple and hoist raise or lower the MCO,
subject to restrictions by the MHM interlocks.

Shielding. The MHM shielding is part of the cask and turret system and is designed in
accordance with the requirements of W-379-C-CSB-13091, CSB MCO Radiation Source, Dose
Rate and Material Properties for Shielding Analysis. The dose rate criterion for the MHM is
0.2 mrem/h (2.0 × 10^{-3} mSv/h) at contact. The design objective is based on 10 CFR 835 for
controlling personnel exposure from external sources of radiation in areas of continuous
occupational occupancy (2,000 h/yr). This design objective is to maintain exposure levels below
an average of 0.5 mrem/h (5.0 × 10^{-3} mSv/h) and as far below this average as is reasonably
achievable. The 0.2 mrem/h (2.0 × 10^{-3} mSv/h) at contact criterion considers the possibility of
multiple other sources associated with the CSB and is based on ALARA program evaluations.
Shielding calculations documented in CSB-SH-1004, Bulk Shielding for the MHM and Portable
Shield Gate, demonstrate that shielding for the MHM satisfies these criteria.

Gamma and neutron shielding are integral to the MHM design. The cask and turret system
has full gamma and neutron shielding. The gamma shielding of the cask and turret body is made
from cast steel (ASTM A36/A36M) sections (10-3/4 in. thick) and is rigidly bolted together with
face-to-face contact joints. The horizontal rotating interface gap between the base of the rotating
turret and the top face of the stationary turntable is kept to a practical minimum of 0.0625 in. and
is stepped to provide a minimum metal thickness in any direction comparable to that of the cask and turret body. A 4-in.-thick outer layer of boron-infused, densified wood (Jabroc N\(^2\)) provides neutron shielding where required. Neutron shielding is needed because approximately 30% of the estimated dose is from neutron radiation.

An alpha-beta CAM can monitor the MHM's cask and turret cavity at the ventilation system's HEPA filter.

The slant-angle concrete path and the gap between the MHM and the deck surface need to be taken into account when estimating the dose rate at the floor surface of the operating deck. The dose rate, including streaming, should be less than the 0.2 mrem/h \((2.0 \times 10^{-3} \text{ mSv/h})\) criterion. The dose rate is reduced by a shield skirt that extends at least 36 in. beyond the edge of the floor embed that surrounds the storage tube. Based on calculation CSB-SH-2002 this distance is judged to be adequate. To complete the gamma shielding at charge face level, the retractable shield skirt is lowered to rest on the deck. The lower turntable at the bottom end of the rotating turret provides protection from gamma radiation coming from the MCOs in the storage tube. (Approximately 5 in. of steel effectively blocks the path, in addition to the very narrow and extended gap based on the extended skirt.)

The nonrotating lower turntable also is manufactured from cast steel and, in conjunction with the components of the retractable shield skirt, has a net shielding thickness comparable to that of the cask and turret body. At the turntable and around the retractable shield skirt, the neutron shielding is provided by cast concrete. Elsewhere, neutron shielding is boron-infused densified wood. A circular slab forming part of the retractable shield skirt functions as a gamma gate when the cask and turret system is rotated to the "camera" index position during intermediate operations and during traveling operations. The thickness of the MCO hoist enclosure and the top plate is governed by the gamma shielding required.

A retractable shield skirt attached around the lower end of the MHM base can be lowered to the operating deck to provide continuous shielding whenever an MCO is engaged. The body of the skirt is designed to provide radiation shielding at the interface. With the skirt raised, there is a 2-in. nominal clearance between the bottom of the skirt and the operating deck. The retractable shield skirt is raised and lowered with motor-driven screw jacks. The mode of suspension has sufficient freedom of motion to ensure that vertical potential seismic displacements are accommodated and that the shielded cask and turret system does not hammer the operating deck. Torque limiters and associated interlocks are provided to avoid overtorquing the drive motors if the skirt-deck interface malfunctions.

**Maintenance.** After supervisory approval, the MHM will move to the MHM maintenance pit, where the maintenance pit equipment frame is located, for supporting the retractable shield skirt and retractable nose for maintenance. The MHM will be positioned over the MHM

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\(^2\)Jabroc N is a product of Permalli Gloucester Limited, Gloucester, England.
maintenance pit equipment frame and aligned with the assistance of a spotter in the pit providing
direction to the operator. The maintenance pit equipment frame has the necessary interface
components to remove the MHM retractable nose assembly.

The MHM maintenance trolley sets on the concrete operations area deck during assembly
and maintenance disassembly operations. The maintenance trolley enables disassembly of the
MHM and provides tooling and fixtures for major repairs. The MHM maintenance trolley is a
rigid structure with dual casters in each corner. It is rolled into position over the maintenance pit
and centered on the maintenance pit equipment frame. As components are removed, the
maintenance trolley supports and maintains stability for the equipment above. Additional frames
and structures are provided with the MHM maintenance trolley to perform different tasks as
needed.

Also located in the MHM maintenance pit is the tube plug and impact absorber exchange
station (Figure A2-9, sheet 8). This station provides a below-floor-level area where the MHM
can deposit or retrieve a tube plug or an impact absorber for changeout, inspection, maintenance,
repair, or surveillance. The top of the tube plug and impact absorber exchange station consists of
flat steel deck plates flush with the load-in/load-out area deck. Beneath the steel deck plates is
the support steel framework, which supports the steel deck plates. The support framework also
supports the MHM shield skirt when the MHM is in position for exchanging a tube plug or an
impact absorber. The tube plug and impact absorber exchange fixture is positioned below the
steel deck plates and rests on the MHM maintenance pit floor station (Figure A2-9, sheet 8).
The tube plug and impact absorber exchange fixture is independent of the support steel
framework and steel deck plates. The steel deck plates are connected to the support steel
framework and anchored to the MHM maintenance pit walls. A removable steel hatch plate
covering the access opening in the deck plates allows the MHM to gain access to the tube plug
and impact absorber exchange fixture. A tube plug support ring for a tube plug or an impact
absorber insert subassembly for an impact absorber is mounted on the tube plug and impact
absorber fixture as needed. The support ring or insert subassembly holds the tube plug or impact
absorber in place during MHM depositing and retrieving operations.

Safety Considerations. The MHM seismic restraints and the MHM rails and rail frogs are
classified as safety class and important-to-safety Category A. The lower flanges of the standard
and overpack storage tubes, the interface guide ring funnel, and the MHM MCO centering guide
are classified as safety significant. The MHM interlock that prevents radiation exposure and
translational shear (P2); the MHM interlock that prevents turret rotation when the MCO hoist is
not fully raised (P6); the MHM interlock that prevents hoist operation and turret rotation unless
the seismic clamps are applied (P21); the MCO hoist and grapple design; and MHM fixed
shielding are safety significant defense in depth. The MHM structural components and the MCO
hoist and grapple are safety significant and important-to-safety Category B. The remaining
components of the MHM, the collision avoidance system, the seismic detection and MHM power-
disconnect system, and the anticollision system are classified general service.
A2.5.1.4 Multi-Canister Overpack Sampling/Weld Operations. The MCO sampling system (Figures A2-40 through A2-46) is used for withdrawing a sample of gases from the monitored MCOs according to a predetermined schedule. Weld equipment will provide additional sealing of the MCO by welding a cap on the MCO.

A2.5.1.4.1 Sampling Operations. MCOs designated as monitored MCOs are sampled periodically at a sampling/weld station to provide data on interim storage of MCOs containing SNF. The sampling procedure (based on HNF-SD-SNF-CN-027, SNF Process Validation Requirements) provides a process for monitoring MCO storage of SNF at the CSB. Several (one to six) MCOs, with and without scrap baskets, will be selected from each K Basin for the MCO monitoring program. The monitoring program will attempt to provide as much information as possible from monitoring selected MCOs. For this reason, the monitored MCOs may include SNF with high and/or low aluminum hydroxide concentrations, and MCO's loaded for specific plutonium blending and single pass reactor fuel. The analytical data and MCO pressurization models indicate that the maximum MCO pressure expected after a 40-year storage period would be approximately 62 lb/in² gauge. The data from the monitoring program will be evaluated. If the monitoring program detects a higher pressurization rate, an engineering evaluation will be organized to review the data, make recommendations, and develop an action plan. A record will be maintained identifying the monitored MCOs and will include data on the SNF stored in the MCO, data from the vacuum drying operations, preselected storage tube location, and data collected from pressure and sampling activities during interim storage in a preselected storage tube. Only one MCO will be placed in each preselected storage tube. Upon receipt at the CSB, an MCO selected for monitoring will be placed in a preselected storage tube. The MCO will be transported using the MHM to the sampling station in the sampling/weld area for measurements of temperature, internal pressure, and gas composition. The MCO pressure will be checked and the MCO will be inerted to approximately 7 lb/in² gauge if needed, and the MCO will be replaced into the preselected storage tube. The monitored MCOs will again be pulled from the preselected storage tubes and checked and sampled during the winter months and summer months. These sampling checks will provide a history (approximately 2 years) of internal gas generation rates and pressure buildup. After sampling has been completed, the MCO cap will be welded on the MCO, and the MCO will be placed in a standard storage tube for interim storage. The SNF Project is considering implementing a long-term MCO pressure monitoring program (port penetration is not required) to detect pressure in excess of the MCO design pressure. The MCO pressure will be transmitted by an internal device to an external receiving device.

Major Components. The major components of the MCO sampling operations are as follows:

- Sampling/weld station impact absorber
- Sampling/weld gantry cranes and hoists
- Sample cart
- Sample hood
- Sampling station cooling cap
- Sampling station chiller
Sampling station HEPA filter and exhauster
Center shield plate
Shield halves
Sampling/weld station MCO support structure
Trench cover shields
Drive assembly cover shields.

Equipment Description and Operational Considerations. The sampling process begins when the MHM lowers a monitored MCO into the sampling/weld pit. The sampling/weld station shielding (shield halves, trench cover shields, drive assembly cover shields, and center shield plate) are used whenever a monitored MCO is raised or lowered into the sampling station to reduce the radiological streaming coming from the MCO to within ALARA limits. The shield halves also form an inverted truncated cone opening, which mitigates accidental dropping of an MCO into the sampling/weld pit. The rotating shielding and fixed shielding in the sampling/weld pits (Figure A2-40) are a composite steel and borated-polyethylene laminate designed to limit gamma and neutron radiation from the MCO to less than 10 mrem/h according to CSB-SH-3005, Sample Station Shielding Calculation. An MCO placed in the sampling/weld pit is supported by the impact absorber on the bottom of the rotating shield which rotates on bearings retained in the stationary shielding (Figure A2-40). The weight of the rotating shield and the MCO is supported by the stationary shielding. The weight of the stationary shielding is supported by six 6-in. diameter pipes which are supported by a carbon steel plate on the sampling/weld station pit floor. A shear ring, which reduces radiation streaming from the MCO, is attached to the rotating shield using 12 screws. The shear ring is designed to shear if an MCO is accidentally dropped into the sampling/weld station. Damage to an MCO is mitigated by the impact absorber. The sampling/weld station features that could be challenged, and whose failure could lead to MCO damage, are designed to safety-class seismic (0.35 g) spectra. The fixed optical pyrometer senses and measures, and the distributed control system (DCS) records, the MCO profile skin temperature as the MCO is lowered into the sampling station. An impact absorber in the bottom of the rotating shield will cushion the MCO if it is dropped and prevent breach of the MCO. The DCS skin temperature data can be used to indicate possible heatup problems from the SNF fuel in the MCO. After placing the MCO in the pit, an optical pyrometer (hand-held) is used to measure the MCO cover surface temperature. An MCO cover surface temperature exceeding a predetermined range will require cooling of the cover before sampling operations begin and protects personnel from skin burns. A chilled cooling cap will be placed on the MCO cover to cool the top surface to a safe temperature range (typically less than 120 °F). Cooling the MCO also maintains MCO temperatures within the safe limits of less than 180 °F delta between the MCO shield plug assembly and the MCO shell temperature, and less than 180 °F/h heat-up/cool-down rate (HNF-SD-SNF-SARR-005).

The sampling/weld station impact absorber assembly (Figure A2-40, sheet 4) consists of a top plate, a base plate, and 13 to 17 energy absorbing tubes between the top plate and base plate. The top plate and base plate have 0.25-in.-deep grooves providing alignment and a piloting surface for each tube component. Carbon steel was selected for the tube material because of its ductility and energy absorbing capabilities. The three sampling/weld station impact absorber
design features selected to minimize deceleration forces of an accidentally dropped MCO are identical to the three design features selected for the bottom and intermediate impact absorbers discussed in Section A2.4.3.1. Approximately half of the crush tubes are shorter by half a crush wave length and symmetrically arranged as shown in Figure A2-40, sheet 4. The three galvanized steel cables mounted circumferentially around the assembly hold the components together and, in case of an accidentally dropped MCO, remain undamaged and hold the components together during removal of the damaged impact absorber. The number of tubes selected for each sampling/weld station impact absorber will be based on full-scale impact absorber tests as discussed in Section A2.4.3.1.

The MCO temperature and pressure are recorded; and gases in the MCO are sampled using the MCO sample cart. An analysis of the gases determines the hydrogen, oxygen, radioactive particulate, and radioactive gas concentrations. After monitoring and sampling operations are complete, the MCO pressure is reestablished with inert gas. The MCO mechanical seal and cover plate may be leak tested to confirm seal integrity. These monitored MCOs will eventually be transferred to a weld station for the cover cap welding operation. If required, the sampling station chilled cooling cap is connected to the chilled supply and return lines to cool the MCO cover. The stationary shielding and cooling cap are designed to cool the MCO by removing heat from the MCO body and top (Figure A2-40, sheet 2).

The sampling/weld station shield halves, trench cover shield, trench grates, and center shield plate are removed and stored, and guard rails are installed to prevent workers from falling into the open trench and the sampling/weld station pit. A ventilation supply system has not been provided for the sampling/weld trenches because the trenches are shallow. If the MHM inadvertently moves near the sampling/weld gantry crane, a crash shield guard attached to the sampling/weld gantry crane (Figure A2-41) will activate the MHM collision avoidance system to prevent a collision of the MHM with the sampling/weld station sample hood. The gantry crane also has seismic restraints to prevent horizontal motion of the crane and the hood. Personnel use the rotating shielding, driven by a worm-gear drive motor (Figure A2-40), to rotate the MCO sample port to an accessible position for sampling operations. Rotating the MCO reduces the radiation dose received by personnel leaning over the MCO. The MCO port cover plate is removed, and the MCO valve operator (Figure A2-40, sheet 3) is bolted to the MCO. The 1.0-ton crane hoist is used to lower the sampling hood over the sampling pit. The sampling hood is used to confine potential airborne contamination coming from an accidental MCO release during sampling operations and dilute any hydrogen released.

The sampling cart is moved near to the sampling pit, and connections are made to the utilities in sampling/weld pit. The quick-disconnect lines from the sampling cart are attached to the inert gas supply, the sampling hood filter, and the discharge line of the sampling hood. The sampling hood flexible exhaust line is connected to the sampling station HEPA filter and exhauster (CSB-AH-006). The exhauster provides a negative sampling hood pressure relative to the sampling/weld area, maintains air contamination control around the sampling pit, dilutes the gases released, and protects operating personnel from any MCO gases that might escape during sampling operations. A negative pressure in the sampling hood and pressure testing of the
sampling lines connected to the MCO provide for safe sampling operations and prevent
contamination of the sampling/weld area and a potential hydrogen deflagration. The exhauster
airflow ensures that any openings in the sampling hood have an air velocity of greater than
100 linear ft/min into the hood to prevent contamination in the hood from escaping into the
operating area. Flow rate and pressure are controlled using load and duct dampers.

Once the sample hood exhaust is functioning, the sample cart lines are purged with inert
gas to remove air, which has the potential for mixing with MCO gases and forming a flammable
mixture. The sample cart is connected, pressure checked, and ready to sample the MCO.

Personnel use the MCO process valve operator to open the MCO process valve. Gases
from inside the MCO flow to the sample cart pressure gauge. The MCO pressure is read and
recorded. The MCO's internal pressure reading is taken through the sample cart system; if it falls
within a specific pressure range, it is sampled and re-inerted. The collected gas sample is sent to
the laboratory for gas analysis. The recorded temperature and pressure readings and gas analysis
sample results provide the data to accurately calculate the gaseous contents of the MCO and
develop a gas composition history for each monitored MCO.

The sampling hood and HEPA filter (FH-9) provide a confinement barrier preventing the
release of particulate contamination to the operating area environment.

After sampling operations, the sample cart pipelines are vented and purged. The purge
allows inert gases to mix with the exhausted MCO gases and dilute the MCO gases before
discharging through the sampling station HEPA filter and exhauster and into the exhaust vent
ducting to the support area exhaust stack. The purge flow is designed to prevent a potentially
flammable gas mixture from entering the sampling station HEPA filter and exhauster.

Opening the MCO process valve, the surge valve, and the refill valve reinerts the MCO
(Figure A2-42). A pressure relief valve located downstream of the MCO vent gas connection
prevents overpressurization of the MCO sampling system. The MCO pressure can be verified by
opening the PI-736 toggle valve, reading the pressure on PI-736, and comparing with PIT-721.
Opening the refill valve during pressure release of the MCO ensures that inert gas dilutes the
released hydrogen to safe concentrations below the flammability limits. The refill valve is allowed
to remain open for a few more seconds to purge the vent line. The MCO process valve operator
is closed and removed, and the remaining sample cart lines are purged by opening the refill valve
and the surge valve to allow inert gases to flow into the sample hood. Closing the surge valve
prevents air from entering the sample cart lines and accumulator. The sampling station HEPA
filtered exhauster may be shut down, the sampling cart lines are disconnected, and the sample
hood flexible line to the HEPA filter and exhauster is removed. The sample hood can now be
lifted off of the sampling station pit using one of the 1.0-ton hoists.
The MCO process port cover leak testing equipment comprises a cover that self-seals around the MCO cover, a vacuum pump that pulls a vacuum on the MCO dome cover, and a mass spectrometer for determining the helium leakage rate. The examiner uses this equipment to sense and record the leak rate for the MCO upper seal.

The guard rails are removed and the sampling/weld station shielding is reinstalled. The sample station chiller is shut down. The MHM then removes the monitored MCO from the sampling station and places it in the assigned preselected storage tube. The sampling pit cover plate is placed over the sampling/weld pit if no other monitored MCOs are to be placed in the sampling/weld pit.

Table A2-5 lists the equipment and operating conditions for which the sampling/weld station has been designed.

Safety Considerations. The sampling/weld station impact absorber, shield halves, and helium rupture disk PSE-1 are classified as safety significant. The helium rupture disk is important-to-safety Category B. Some components of the MCO sampling equipment (the sampling/weld station MCO support structure [which includes the rotating shield, the stationary shielding, the shear ring and shear screws, and the stationary shield support pipes], sample hood and HEPA filter, exhaust system [HEPA filter], sample hood exhaust flow indicator, center shield plate, and MCO valve operator) are classified safety significant. Because a potential drop of a shield half or sampling hood could cause minor damage to the top of an MCO, the sampling/weld gantry crane structure and hoist are classified important-to-safety Category C (Figure A2-41).

A2.5.1.4.2 Canister Cover Assembly Welding Operations. MCOs received at the CSB have a canister cover assembly welded to them before being placed for interim storage in the standard storage tubes. Monitored MCOs received at the CSB are placed in a standard storage tube and periodically sampled at a sampling/weld station. At the completion of monitoring activities, a canister cover assembly is welded to the MCO, and the MCO is placed in a standard storage tube. The canister cover assembly together with the stainless steel shell and bottom provide a container for the SNF (Figure A2-2) that meets all regulatory requirements for SNF dry storage (HNF-SD-SNF-SARR-005). The canister cover assemblies are welded using the welding equipment in pit numbers 2 and 7 in the sampling/weld area of the operating area shelter (Figure A2-43). MCOs received from the CVDF are transported by the MHM to these pits, and welding equipment, attached to the cover assembly, welds the canister cover assembly to the MCO. A connector installed on the south wall provides inert weld cover gases through a hose to the welding equipment.
<table>
<thead>
<tr>
<th>Component</th>
<th>System design conditions</th>
<th>System operating conditions</th>
<th>Design operating environment</th>
<th>Operating environment</th>
<th>Material</th>
<th>Corrosion allowance</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sampling/weld station chiller (CHW-001)</td>
<td>3°F to 115°F</td>
<td>3°F to 115°F</td>
<td>3°F to 115°F</td>
<td>-25°F to 115°F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td>Sampling/weld station HEPA filter and exhauster (AH-006)</td>
<td>60°F to 104°F</td>
<td>0 to -0.5 w.g.</td>
<td>32°F to 400°F</td>
<td>60°F to 85°F</td>
<td>304L stainless steel</td>
<td>NA</td>
</tr>
<tr>
<td>Sample flow indicating device</td>
<td>150 lb/in² gauge</td>
<td>0 to -0.5 w.g.</td>
<td>32°F to 400°F</td>
<td>60°F to 85°F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td>Sampling/weld station impact absorber (IMP-005)</td>
<td>60°F to 104°F</td>
<td>60°F to 85°F</td>
<td>32°F to 130°F</td>
<td>60°F to 85°F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td>Sampling/weld station gantry cranes (CRN-009/HOL-710)</td>
<td>Travel speed: 35 ft/min</td>
<td>Lift speed: 8 ft/min</td>
<td>Trolley speed: (5 ton) 25-ft/min</td>
<td>Up to design conditions</td>
<td>32°F to 104°F</td>
<td>60°F to 85°F</td>
</tr>
<tr>
<td>Sampling/weld station auxiliary hoist (HOI-711)</td>
<td>Lift speed: 8 ft/min</td>
<td>NA</td>
<td>32°F to 104°F</td>
<td>60°F to 85°F</td>
<td>NA</td>
<td>NA</td>
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<tr>
<td>Sampling/weld station auxiliary hoist (HOI-712)</td>
<td>Lift speed: 8 ft/min</td>
<td>NA</td>
<td>32°F to 104°F</td>
<td>60°F to 85°F</td>
<td>NA</td>
<td>NA</td>
</tr>
<tr>
<td>MCO sample cart (Cart-001)</td>
<td>150 lb/in² gauge</td>
<td>0 to 7 lb/in² gauge</td>
<td>32°F to 130°F</td>
<td>60°F to 85°F</td>
<td>Stainless steel piping</td>
<td>NA</td>
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<tr>
<td>12 in. pipe (REC-3)</td>
<td>150 lb/in² gauge</td>
<td>0 to 7 lb/in² gauge</td>
<td>32°F to 130°F</td>
<td>60°F to 85°F</td>
<td>304L stainless steel, ASME VIII</td>
<td>NA</td>
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<tr>
<td>Sample hood (BARR-002)</td>
<td>60°F to 104°F</td>
<td>60°F to 85°F</td>
<td>32°F to 104°F</td>
<td>60°F to 85°F</td>
<td>Stainless steel, polycarbonate</td>
<td>NA</td>
</tr>
<tr>
<td>Sample hood HEPA filter (FH-9)</td>
<td>150 lb/in²</td>
<td>0 to 7 lb/in² gauge</td>
<td>32°F to 400°F</td>
<td>60°F to 85°F</td>
<td>316L stainless steel</td>
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<tr>
<td>Sample hood hose assembly</td>
<td>150 lb/in²</td>
<td>0 to 7 lb/in² gauge</td>
<td>32°F to 130°F</td>
<td>60°F to 85°F</td>
<td>Carbon steel</td>
<td>NA</td>
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<td>Sampling/weld station cooling cap (HX-002)</td>
<td>3°F to 115°F</td>
<td>3°F to 115°F</td>
<td>32°F to 130°F</td>
<td>60°F to 85°F</td>
<td>Carbon steel</td>
<td>NA</td>
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<td>Weld hoods (BARR-003/BARR-004)</td>
<td>Hydrostatic (burst) 450 lb/in²</td>
<td>Atmospheric pressure</td>
<td>32°F to 104°F</td>
<td>60°F to 85°F</td>
<td>Stainless steel, laminated vinyl/nylon</td>
<td>NA</td>
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<td>Gantry crane power carrier (PWC-001/PWC-002)</td>
<td>60°F to 104°F</td>
<td>60°F to 85°F</td>
<td>32°F to 130°F</td>
<td>60°F to 85°F</td>
<td>Carbon steel</td>
<td>NA</td>
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<tr>
<td>Cooling cap storage tray (RCK-001)</td>
<td>60°F to 104°F</td>
<td>60°F to 85°F</td>
<td>32°F to 130°F</td>
<td>60°F to 85°F</td>
<td>Carbon steel</td>
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<td>Portable guard rail (GR-001A/B)</td>
<td>NA</td>
<td>NA</td>
<td>32°F to 130°F</td>
<td>60°F to 85°F</td>
<td>Carbon steel</td>
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<td>Removable guard rail (GR-002A/B)</td>
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<td>NA</td>
<td>32°F to 130°F</td>
<td>60°F to 85°F</td>
<td>Aluminum</td>
<td>NA</td>
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<tr>
<td>Component</td>
<td>System design conditions</td>
<td>System operating conditions</td>
<td>Design operating environment</td>
<td>Operating environment</td>
<td>Material</td>
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<td>60 °F to 85 °F</td>
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<td>Removable guard rail (GR-005A/B)</td>
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<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Aluminum</td>
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<tr>
<td>Railing support brackets (BKT-001A/B)</td>
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<td>NA</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel</td>
<td>NA</td>
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<td>Shield support stand (SUP-003)</td>
<td>NA</td>
<td>NA</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel</td>
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<td>Shield plug (PL-004)</td>
<td>NA</td>
<td>NA</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel</td>
<td>NA</td>
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<td>Shield plug grapple (GPL-003)</td>
<td>NA</td>
<td>NA</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
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<td>Rotating shield (RSE-005)</td>
<td>NA</td>
<td>NA</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
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<td>Stationary shield (RSE-006)</td>
<td>NA</td>
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<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel</td>
<td>NA</td>
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<tr>
<td>Shield half (North) (RSE-007)</td>
<td>NA</td>
<td>NA</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td>Shield half (South) (RSE-008)</td>
<td>NA</td>
<td>NA</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td>Trench cover shield (COV-005, COV-006)</td>
<td>NA</td>
<td>NA</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td>Drive assembly cover shield (COV-007, COV-008)</td>
<td>NA</td>
<td>NA</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td>Shield lifting sling (COV-006)</td>
<td>NA</td>
<td>NA</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td>MCO rotation drive system (DRV-001)</td>
<td>NA</td>
<td>NA</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td>Sample cart interface panel</td>
<td>NA</td>
<td>NA</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td>480 volt panelboard (DA-33-217)</td>
<td>NA</td>
<td>NA</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td>Transformer (XT-33-218)</td>
<td>NA</td>
<td>NA</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>NA</td>
<td>NA</td>
</tr>
<tr>
<td>120 volt panelboard (DA-33-307)</td>
<td>NA</td>
<td>NA</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>NA</td>
<td>NA</td>
</tr>
</tbody>
</table>

This table is for information only and does not provide operational controls.


HEFA = high-efficiency particulate air (filter).
MCO = multi-canister overpack.
NA = not applicable.
**Major Components.** The following major components are part of the canister cover assembly welding operations (in addition to the equipment provided for sampling operations):

- East sampling/weld station gantry crane and hoists
- West sampling/weld station gantry crane and hoists
- Weld hood
- Welding equipment
- Inert gas manifold.

**Equipment Description and Operational Considerations.** Before welding operations begin, the following preliminary steps are completed: (1) verify that the sampling/weld station impact absorber is in place and in good condition, (2) verify that the sampling/weld station shield halves, trench cover shield, and drive assembly cover shield are in place, (3) verify that the center shield plate is in place, and (4) verify that the sampling/weld station gantry crane and auxiliary equipment have been moved away from the sampling/weld station pit where the canister cover assembly will be welded to the MCO. These steps prepare the sampling/weld station for access by the MHM for lowering of an MCO into the sampling/weld station pit.

The sampling/weld area has seven pits with pits 2 and 7 each equipped with a sampling/weld gantry crane, as described in Section A2.5.1.4.1. Both pits 2 and 7 are equipped for welding operations, and pit 7 is equipped for MCO monitoring operations. Pits 2 and 7 are approximately 40 ft apart. The east and west sampling/weld gantry cranes use the same east–west wheel tracks. Anticollision devices prevent the east and west sampling/weld gantry cranes from colliding. Travel for the east sampling/weld gantry crane is physically limited to pits 4 through 7, and travel for the west sampling/weld gantry crane is limited to pits 1 through 4. If an inadvertent collision between these two cranes occurred, there would be no adverse consequences because the collision would occur over pit 4, which would not contain an MCO, and the cranes would not have hoods connected to MCOs in pits 2 or 7. Each gantry crane has a crash shield guard to prevent a collision with the MHM.

The MHM transports the MCO to the sampling/weld station from the cask receiving pit or transports a monitored MCO from a storage tube. After centering the MHM over the sampling/weld pit, the retractable nose unit and shield skirt are lowered, the center shield plate is retrieved into the MHM plug cavity, and the MHM lowers an MCO into the sampling/weld station and replaces the center shield plate. The sampling/weld station shield halves, trench cover shield, and drive assembly cover shield in contact with the MHM retractable nose unit and shield skirt provide shielding for personnel from the MCO while it is being lowered. As the MCO is lowered into the sampling/weld station pit, an infrared pyrometer, TI-723, monitors the MCO surface temperature. These temperature readings are recorded and tracked on the DCS.

After the MHM has moved away from the sampling/weld station, personnel use the sampling/weld station gantry crane hoist (Figure A2-44) to remove the center shield plate and the shield halves and place them in their temporary storage places. Personnel then install the removable handrails and removes the sampling/weld station pit covers. These covers are also
placed in their temporary storage locations as shown in Figure A2-43. The removable guard rails, after being installed into position, provide protection for personnel from an accidental fall into the sampling/weld station pit or into the pit trench.

Removal of the shield halves, trench cover shield, and trench grates exposes the top of the MCO's canister collar for welding operations. Personnel descend into the pit trench and use a hand-held infrared pyrometer to measure the temperature of the MCO. At MCO shield plug temperatures within the predetermined safe range (typically less than 120 °F), personnel can work safely near the MCO. If the MCO temperature is higher than the safe range, personnel place a portable cooling cap on the MCO, and a chilled water flow circulating around the sampling/weld station stationary shielding and through the portable cooling cap cools the MCO. Both the portable cooling cap and the stationary shielding have supply and return lines connected to the sampling/weld station chiller unit. The sampling/weld station chiller is mounted on a concrete pad outside of the operating shelter on the south side (Figure A2-45). The sampling/weld gantry crane's 1-ton hoist will be used to place the portable cooling cap on the MCO shield plug assembly if it is needed.

After ensuring that the MCO shield plug temperature is within a safe operating temperature range, a canister cover assembly is retrieved from storage, affixed to the MCO weld hood lifting clamps, and positioned over the MCO using the 1-ton auxiliary hoist of the sampling/weld station gantry crane. Welder set-up time can be reduced by placing the welder assembly on the canister cover assembly in the storage area, positioning the welding track to circumvent the weld area, and clamping the welder to the canister cover assembly. After the canister cover assembly with the attached welder assembly has been placed on the MCO, only minor welder assembly adjustments should be required before the automatic welding process begins. Once the canister cover assembly is in place, the ventilation fan for the weld hood is turned on, the ventilation flow dampers are positioned to provide an adequate airflow rate, and the HEPA filter differential pressure is checked. These steps provide assurance that only gases generated by the weld process (radioactive or hazardous) will be extracted by the ventilation system from the weld area. The gas tungsten arc welding process only emits trace amounts of noxious gases mixed with the inert gas; however, minute amounts of residual contamination may adhere to the outer surface of the MCO and become airborne during the welding process.

The canister cover assembly weighs approximately 500 lb. The weld area on the MCO is examined for cleanliness before the canister cover assembly is placed on the MCO. The procedures for attaching, lifting, and moving the canister cover assembly contain safety precautions for personnel. A weld technician correctly positions the canister cover assembly and checks the weld area for alignment and cleanliness. The weld technician checks that the utility supply lines (argon, and electrical power and instrument cables) are connected to the welding equipment, the welding head is positioned correctly, and the welding wire supply is adequate, and also checks the controls at the welding machine remote operating terminal (Figure A2-46). The weld head is used to temporarily attach the canister cover assembly to the MCO using several tack welds. After the tack welding operations are completed, the canister cover assembly is welded to the MCO using a number of passes. Two video cameras mounted on the weld head
monitor and record the welding process. A television monitor next to the welding machine’s remote-operating terminal allows the weld technician to view and control the welding process. The video recording can be used as part of the verification process used to certify weld quality. The weld area is allowed to cool before checking for weld quality. Weld verification and certification also includes a visual examination of the cooled weld area for flaws and other weld quality examination procedures (i.e., liquid-penetrant examination and helium leak checking). The inert gas supply system summarized in Section A2.7.5 provides helium for leak testing of the canister cover assembly. The welding and examination procedures ensure that the MCO canister cover assembly weld meets the requirements of the ASME Boiler and Pressure Vessel Code, Section III, subsection NB (ASME 1995).

After a successful welding operation, the ventilation damper is closed shutting off airflow. The welding equipment is disengaged from the canister cover assembly, the sampling/weld station chiller valves are closed, and the weld hood is lifted off of the MCO and returned to its storage location. The trench cover shield and trench grates are replaced, and the removable guardrails are removed. The sampling/weld station gantry crane’s 5-ton hoist is used to reinstall the shield halves and the center shield plate. The affected gantry crane is moved away from the sampling/weld station pit, and the MHM retrieves the capped MCO and transfers it to a storage tube. Simultaneous welding operations can occur in both pits 2 and 7, or simultaneous welding and sampling operations can occur in pits 2 and 7.

The air-cooled chiller is a factory-assembled refrigeration unit designed to provide and circulate a 40% to 50% propylene glycol mixture for both pits. The unit is pad mounted outside of the operating shelter on the south side. Insulated pipelines enter the operating shelter at just above ground level and extend into sampling/weld station pits 2 and 7 through encased pipes under the deck (see Figure A2-44). The chilled water cools the sampling/weld station stationary shielding by flowing through pipes in contact with the outer surface of the shielding. Flexible and insulated connecting hoses can also be connected to the portable cooling cap if separate cooling of the MCO top is needed.

Components of the air-cooled chiller unit include an evaporator, storage or expansion tank, circulating pump, controls, refrigerant piping, and accessories. The unit has been designed with weatherproof protection for outdoor use. A fence surrounding the chiller unit protects the chiller components from passing traffic. The chiller has been designed to have a net cooling capacity of 60,000 BTU/h and to meet the standards, rating, and testing requirements of ANSI/ARI 590, Positive Displacement Compressor Water Chilling Package.

An inert gas manifold assembly, installed in the argon gas manifold building, provides cover gases for the welding process to prevent oxidation of the welded area. The argon gas manifold building, located outside and adjacent to the chiller unit (Figure A2-45), contains a gas manifold cylinder rack and a cabinet providing storage for spare and used argon high-pressure gas cylinders. The argon gas manifold building is a steel-frame building with corrosion-resistant metal siding and roof. The roof is attached using corrosion-resistant fasteners. The cylinders are maintained in an upright position by chaining them to the cylinder racks. All pressurized gas
cylinders received at the CSB have a cylinder cap that remains on the cylinder until after the
cylinder is chained to the cylinder rack. All cylinders have a cylinder valve. A maximum hole size
in the inlet to the butt of the cylinder valve restricts airflow through the valve and minimizes
propulsion effects should the valve be accidentally severed at the top of the cylinder. Both the
cylinder cap and the cylinder valve protect the cylinder from causing serious propulsion accidents.

Pipelines from the argon gas manifold building enter the south side of the operating shelter above
floor level and terminate just inside the south wall with valves and quick-disconnect couplings
(Figure A2-45 is a flow diagram of the chilled glycol system). The valves and quick disconnect
couplings are located in the sampling/weld area opposite sampling/weld station pits 2 and 7.

Safety Considerations. The sampling/weld station impact absorber and shield halves are
classified as safety significant. Some components of the canister cover assembly welding
equipment (the sampling/weld station MCO support structure and center shield plate) are
classified safety significant, and the remainder of the components are general service. The
sampling/weld gantry crane structure and hoist are important-to-safety Category C because the
crane could potentially drop a shield half or sampling hood causing minor damage to the top of an
MCO.

A2.5.2 Passive Cooling of the Canister Storage Building Storage Vault

The CSB contains an array of storage tubes that are cooled using passive, naturally
circulating air to remove the decay heat from the SNF (CSB-HV-0001, Table 3) contained in
MCOs in the storage vault. According to HNF-SD-SNF-CN-005, Bounding Temperatures and
Gas Generation Rates during CSB Staging, this system is designed to ensure that the fuel
centerline temperature limit of 600 °C (1,112 °F) and the concrete temperature limit of 65 °C
(150 °F) per the requirements of ANSI/ACI 349-85 are not exceeded.

Major Components. The following major components are part of the passive cooling
system:

- Intake plenum
- Outlet plenum
- Intake stack
- Exhaust stack
- Vault 1
- DCS.

Equipment Description and Operational Considerations. The storage vault is passively
cooled using the "stack effect" to promote ventilation flow. Natural convection occurs when a
heated surface is in contact with fluid (air) of a different temperature than the surface. Density
differences provide the motive energy to move the air. The height difference between the
operating vault deck and the air exhaust stack (169 ft above the operating vault deck) increases
the motive force for airflow (stack effect). Also, the top of the inlet plenum opening is 7 ft lower
than the top of the exit plenum, causing heated air to preferentially expand into the exit plenum and exhaust up the stack. Air passes down through the air intake stack, where its temperature and flow rate are monitored, to an inlet plenum. From the inlet plenum the air flows into the enclosed volume of the vault where it removes heat from the MCO storage tubes. The heated air flows out of the vault into the outlet plenum, up the exhaust stack, where the temperature is monitored, and discharges into the atmosphere. The air intake cruciform and canopy improves airflow efficiencies by reducing wind velocity from any direction to positive pressure. A wind deflector on top of the exhaust stack improves airflow efficiencies for the exiting hot air by creating a negative pressure.

Modeling done using the PHOENICS 2D Computational Fluid Dynamics model, described in PHOENICS Computer Program (CHAM 1991), shows that the temperature difference between the highest vault air temperature (hot spot) and the outlet air temperature is only 9 °F (CSB-HV-0001). According to Conceptual Design Report for Line Item Project Nuclear Materials Storage Facility Renovation at the Los Alamos National Laboratory, Technical Area 55, Building PF-41 (LANL 1995), a similar conclusion was obtained by the Los Alamos National Laboratory for their vault as a result of three-dimensional modeling using the CFX computer code. The range for MCO external surface temperatures is 241 °F to 252 °F for the four gases analyzed for the storage tubes: argon, helium, nitrogen, and air. The standard storage tubes contain air at atmospheric pressure. For air the estimated maximum MCO temperature is 251.4 °F. Helium was selected as the inert gas for the overpack storage tubes.

The design of the cooling system limits the MCO external temperature to a maximum 270 °F wall temperature to ensure the fuel centerline temperature limit of 1,112 °F and the concrete temperature limit of 150 °F are not exceeded. The thermal analysis (CSB-HV-0001) indicates that the maximum concrete temperature inside the vault is 121 °F. This temperature is below the ANSI/ACI 349-85 limit of 150 °F.

Temperature indicators are not needed in vault 1 because (1) the vault thermal analysis shows that air temperatures in the vault will not vary by more than 11 °F, (2) other approved DOE-managed sites (i.e., Fort St. Vrain) have been approved for operation without vault temperature indicators, and (3) even though air temperature monitoring is not required, the exhaust stack air temperature indicator will provide information that the vault temperatures are within temperatures predicted by the vault thermal analysis.

The passive cooling system contains no moving parts and no components likely to have significant deterioration during the design life of the CSB. The temperature of the vault depends on the outdoor temperature, which has a design range of -27 °F to 115 °F. The temperatures in a tube and in the MCOs contained in it depend on the temperature of the vault at that location and on the heat produced by the MCOs. The vault air temperature varies between 125 °F and 136 °F for a fully loaded vault with a maximum ambient inlet air temperature of 115 °F. If a heat wave occurs at the Hanford Site with an outside air temperature of 115 °F, the calculated maximum equivalent MCO surface temperature would be 230 °F for an MCO located in the fully loaded vault (CSB-HV-0001). The use of thermocouples to identify hot spots in the vault is not practical.
because the small 11 °F variability would not have a significant impact in predicting the MCO surface temperature. The exhaust stack is described in Section A2.4. The upper 57 ft of the stack is constructed of 7-ftouter diameter, 3/8-in.-thick steel plate. At an exhaust temperature of 134 °F, the estimated air flow rate through the exhaust stack is 31,700 ft³/min for the maximum vault loading of 161.2 kW and an inlet air temperature of 115 °F. Air inlet and exhaust temperatures and the flow rate are monitored using the DCS. A mass flowmeter and a temperature element are mounted in the vault cooling system intake stack to measure the flow rate and temperature of the incoming air. At the base of the intake stack, the air flows through the intake plenum into the storage vault and around the storage tubes, absorbing and removing heat from the stored fuel. As the air increases in temperature and expands, it flows through the outlet plenum (preferentially because the top of the exhaust stack is 90 ft higher than the intake stack) and out the exhaust stack, where a temperature monitor measures the temperature of the exhausted air. An exhaust flowmeter is not provided as there is no other airflow than through the intake, vault, and exhaust stack.

The thermal analysis for the CSB vault analyzed natural air convection flow due to nuclear decay heat of stored material inside the vault. Three computer programs based on conservation of mass, momentum, and energy were used to model airflow through the vault; T-DUCT, PHOENICS and LOTUS (CSB-HV-0001). The T-DUCT computer model creates numerical models of duct systems using two basic principles of ventilation; conservation of mass and conservation of energy. The T-DUCT simulated the changing cross-sections of the vault intake stack, inside the vault and the exhaust stack and estimated the pressure losses for airflow through the passive cooling system. The PHOENICS program simulated fluid flow, heat transfer, and turbulence based on the physical properties of air using the finite element approach. For vault flow, the flow is mostly turbulent and the problem is solved using the Navier-Stokes equations representing the conservation of mass, momentum, and energy. A LOTUS spreadsheet was used to model the "Hot Spot" in the vault, the area of the vault where, due to the buoyancy effect and air turbulence, the air temperatures are the highest. This spreadsheet solved a series of nonlinear simultaneous algebraic heat transfer equations using the matrix inversion method.

Design criteria are presented in the thermal analysis (CSB-HV-0001, page 8). The SNF MCO heat load for one vault is 161.2 kW. Thermal conditions in the vault are based on the following conservative assumptions:

- The highest outside air temperature, 115 °F
- The highest total heat load, 161.2 kW
- The highest tube heat load, 1.6 kW, surrounded by the tubes with the same highest heat load
- The worst location of the tube, in a "hot spot" area at the end of the vault
- No positive wind effect.
Although a three-dimensional vault thermal analysis was planned, the PHOENICS 2D Computational Fluid Dynamics model has been used showing that the temperature difference between the highest vault air temperature (hot spot) and the outlet air temperature is only 4 °F (CSB-HV-0001; note Figures A4-2 and A4-3 in Chapter A4.0). The Los Alamos National Laboratory obtained a similar conclusion for their vault using the three-dimensional CFX computer modeling code (LANL 1995).

A number of two-dimensional computational fluid dynamics studies were performed to explain the airflow patterns along the longitudinal side of the vault that resulted in such a low temperature difference. There are two major airflow streams in the vault: the primary stream at the bottom of the vault moving from the entry to the exit, and the secondary stream at the upper part of the vault moving in the opposite direction. The modeling shows that intensive airflow movement takes place in the vault zone where the tubes are located. Also, there is a passive airflow vortex with a central stagnant zone located in the corner below the outlet stack. However, this vortex occurs in the no-tube zone. The airflow pattern developed with two-dimensional computational fluid dynamics modeling was used to identify the free area for calculating pressure loss using the local resistance coefficient presented in Handbook of Hydraulic Resistance (Idelchik 1986, page 608, diagram 12-28). As shown in the thermal analysis (CSB-HV-0001), the pressure loss is negligible because of low air velocity in the vault. Three-dimensional modeling would be capable of creating the airflow pattern more accurately, but the results of that modeling would have to be applied to the C-coefficient developed by Idelchik (1986) for unknown and probably different airflow conditions and still negligible pressure loss. The two-dimensional computational models showed (1) a low temperature difference, (2) a vortex in the no-tube zone, and (3) a low air velocity in the vault. Therefore, the more accurate three-dimensional computational fluid dynamics modeling was not necessary.

The conservative assumption that the analyzed tube with the highest heat load is surrounded by tubes with the same heat load allows us to ignore the thermal radiation effect between tubes having the same surface temperatures. However, the thermal radiation effect between tubes and internal surfaces is calculated precisely. Under these conditions, the MCO external surface temperature is 230 °F. The design of the cooling system limits the MCO external temperature to a maximum 270 °F wall temperature to ensure the fuel centerline temperature limit of 1,112 °F is not exceeded. The thermal analysis (CSB-HV-0001) indicates that with the conservative assumptions noted above, the maximum concrete temperature inside the vault is 125 °F. This is below the ANSI/ACI 349-85 limit of 150 °F. The design of the cooling system does not include MCOs in overpack storage tubes.

The vault thermal analysis (CSB-HV-0001) was based on a total heat input of 161.2 kW. The heat load for each MCO is based on Basin data for the SNF loaded into an MCO (exposure, cooling time, and fuel weight). This analysis calculated a maximum surface temperature of 186 °F for an MCO. It is expected that MCOs in overpack storage tubes will likewise be at or below the maximum calculated MCO surface temperature. Placing MCOs in overpack storage tubes at the air entrance point and limiting them to one MCO per tube provide added assurance that the maximum surface temperature of an MCO will not be exceeded.
Because the overpack storage tubes will contain the heat input associated with only one MCO, the vault thermal analysis will not be invalidated as long as the total heat input for MCOs in standard storage tubes as well as overpack storage tubes does not exceed 161.2 kW.

The PHOENICS computer program simulates fluid flow, heat transfer, and other related phenomena. This simulation is based on mathematical derivation from established physical principles. The power and flexibility of PHOENICS has resulted in numerous applications in the aerospace, nuclear, process, defense, marine, environmental, turbo-machinery, and automotive industries. There are more than 350 installations of PHOENICS throughout the world. This program was selected as a result of a study of other existing codes. CHAM company, the author of PHOENICS, stated that the code “is very well validated and developed from over 10 years of commercial use” (CSB-HV-0001).

Safety Considerations. The intake stack, intake plenum/exhaust plenum, exhaust stack, and vault 1 are classified as safety class and important-to-safety Category A. The DCS is classified general service.

A2.5.3 Overpack Storage Tube Purge Operations

The following paragraphs summarize the recovery actions for MCOs and include the equipment and operations for monitoring and maintaining MCOs in overpack storage tubes. Overpack storage tube operations are designed to safely monitor a suspect leaking MCO or a damaged MCO. An MCO will be placed in an overpack storage tube for up to 1 year until monitoring activities determine the leak rate, if any, of gases escaping from the mechanically sealed MCO or from the damaged MCO. If the leak rate is determined to be acceptable based on the approved action plan developed for the MCO, the MCO will be transported to a sampling/weld pit and a canister cover assembly will be welded to the MCO. This MCO will then be placed in a standard storage tube. If overpack storage tube monitoring activities determine that the MCO leak rates are unacceptable (above $1 \times 10^{-5}$ standard cm$^3$/s·atm), the Unreviewed Safety Question recovery plan may recommend one of the following.

- The MCO may remain in the overpack storage tube for further monitoring.
- The MCO may be transported (with additional analysis) to a suitable location for reworking of the mechanical seals to reduce the leak rate.
- The MCO may be returned to the CVDF and the K Basins for MCO repackaging operations.

Using the overpack storage tube for more than 1 year requires performing an Unreviewed Safety Question evaluation. The overpack storage purge system is designed to monitor and maintain an inert gas environment around any MCO placed in the overpack storage tubes and may be used to
monitor the atmosphere in any of the standard storage tubes. A tube vent and purge cart (Figure A2-47) is used for the entire purging process.

Major Components. The following major components are included with the overpack tube purge equipment:

- Tube vent and purge cart
- Portable hoist
- Floor electrical sockets
- Helium fill station.

Equipment Description and Operational Considerations. Operation of the overpack storage tube purge system begins for an overpack storage tube when an abnormal or accident MCO has been identified and an MCO has been placed in the overpack storage tube by the MHM. The overpack storage tube purge system is designed for venting, purging, and refilling the atmosphere in an overpack storage tube containing an MCO. The primary equipment used for the purging process is the tube vent and purge cart described below. The tube vent and purge cart’s oxygen monitor alarms when the oxygen concentration in the airspace immediately around the cart is less than 19.5% to warn personnel of an oxygen-deficient atmosphere in the unlikely event of helium leaking from the purge system or helium exhausted from a purged overpack storage tube.

The overpack storage tube purge system has three main functions for overpack tubes: tube venting, tube pressure-purging, and tube refilling with inert gas.

- MCO overpack tube venting is required when an MCO stored in an overpack storage tube is to be moved to a standard storage tube.
- Purging of MCO overpack tubes is required after lowering an MCO into the overpack tube and the overpack tube plug has been replaced. Purging may also be required when the sampled gases in an overpack storage tube contain hydrogen.
- MCO overpack tube refilling operations are needed when the overpack tube gas pressure falls (caused by a slight gas leak) below 2 lb/in² gauge.

The tube vent and purge cart includes features to analyze gas from the overpack storage tube, purge generated gas (containing hydrogen) from the tube through a HEPA filter to the operating area atmosphere, and replace the purged gas with an inert gas (e.g., helium). The cart contains an inert gas supply (two gas cylinders), a flexible steel hose fitted for the overpack tube and another flexible steel hose for the standard storage tubes; a CAM unit; a radioactive gas monitor; a hydrogen gas monitor with associated alarms; a sampling connection; a HEPA filter; a gas monitor port heat exchanger; and an oxygen concentration monitor with associated alarms (Figure A2-48). The tube vent and purge cart interfaces with the overpack tube plug connector,
the floor electrical supply, the standard tube plug connector, and the helium filling station. The tube vent and purge cart also has the capability to sample standard storage tubes (Figure A2-14).

The tube vent and purge cart is a battery-powered, rechargeable, utility vehicle. The battery is maintenance free and attached to the cart inside a battery compartment designed to contain any acid that might leak from the battery. This battery design and configuration minimizes loss of battery acid and subsequent hydrogen generation caused by acid reaction with exposed metal components. The cart is designed with a rear bed and tow hitch. The rear bed, or cargo deck, is sized for a platform 4 ft, 7 in. by 9 ft, 7.5 in. The bed is fitted with all of the equipment and systems required to perform the tube venting and purging operations. A flexible steel hose connects the cart's piping header with the tube plug quick-disconnect coupling. The tube vent and purge cart's rated load capacity is specified to be not less than 3,000 lb. The cart can tow a portable floor crane. The floor crane is used for lifting the 425-lb center hatch of the standard or overpack tube plug cover. The portable floor crane is used to remove the standard or overpack tube plug covers in preparation for tube plug removal by the MHM. The standard tube plug cover weighs 1,475 lb, and the overpack tube plug cover weighs 1,805 lb. The portable crane's rated capacity is specified to be 2,000 lb. The cart receives electrical power for recharging its batteries and operating equipment by means of an on-board retractable cable reel (480 V, 3-phase, 4-wire cord with ground). The cord is plugged into the electrical distribution system outlets recessed into the operating deck.

During the time the cart is being used to perform venting and purging, the portion of the connected system that is between the cart-mounted HEPA filter and the tube atmosphere serves the same secondary confinement role as the overpack storage tubes. Under accident conditions, this confinement provides mitigation of radionuclide releases from the tubes. During purging operations, the purge cart is secured in place by wheel chocks adjacent to the storage tube being serviced. The portable crane is used to remove the center hatch and move it to a temporary location convenient to the tube. Removing the quick-disconnect cap on the MCO overpack tube plug, connecting the flexible steel hose from the cart to the plug, and opening the pressure-indicating valve on the cart shows the tube pressure. This pressure can be compared with the pressure indicated by the tube plug local pressure gauge. If these two pressure gauges differ by more than 0.5 lb/in² gauge, both instruments are recalibrated before continuing the purging operation.

The purge cart CAM, hydrogen monitor, and radioactive gas monitoring systems measure radioactivity, hydrogen concentration, and radioactive gas concentration of gases being evacuated or purged from the tube. Gas flow rate to the monitoring equipment is monitored, and if the flow rate to the CAM and radioactive gas monitor drops below the setpoint value (or if the flow to the hydrogen monitor drops below the setpoint value), a low-flow alarm activates alerting personnel that the gas monitoring equipment may not be operating properly. The hydrogen monitor alarms if the gas hydrogen concentration exceeds 1% by volume. The gas temperature is measured and an alarm is activated if the temperature exceeds 125 °F. A high gas temperature could cause thermal damage to the gas monitoring equipment. An inline gas cooler has been designed to cool
gases before they reach the CAM and gas monitors and prevent thermal damage to the gas monitoring equipment.

MCO overpack tube gases are vented through a HEPA filter to remove radioactive contamination before the gases pass into the operating area atmosphere. If there is a breach or leak in the MCO, SNF particulate may be suspended in the overpack tube atmosphere. An HPT will provide periodic radiation monitoring of the FH-3 HEPA filter to determine when filter changeout is needed. Venting continues until the pressure becomes atmospheric pressure (0.0 lb/in\(^2\) gauge). After the overpack tube is depressurized, tube plug seal integrity is checked by applying 35 lb/in\(^2\) gauge pressure to the plug seal space and monitoring the pressure for approximately 20 minutes. According to CSB-PR-0002, Plug Seal Leak Rate and Test, a plug seal pressure drop of greater than 0.5 lb/in\(^2\) gauge in that time means that the plug seals should be inspected and replaced if needed. The pressure gauge has increments of 0.5 lb/in\(^2\) gauge.

Overpack tube venting and purging (if required) includes final pressurization of the overpack tube with inert gas to 3 to 4 lb/in\(^2\). This slight gas pressure remaining in the overpack storage tube helps to prevent air from entering the overpack storage tube. After venting and purging the overpack tube, the flexible connection to the tube plug is disconnected, the quick-disconnect cap is replaced, the tube plug center hatch is replaced using the hoist, the cart is disconnected from the electrical outlet, and the cart is returned to the cart storage area. In the cart storage area, cart preparation activities for future storage tube purging or venting operations include recharging the cart battery, refilling the inert gas bottles, calibrating instruments (if needed), and general cart maintenance.

The overpack tube plugs have two interconnected pipe lines for inerting and purging the overpack tube. One line connects to a pressure gauge and the other to a quick-disconnect coupling. The standard storage tubes and preselected storage tubes will not need to be inerted. However, the tube vent and purge cart could be used to monitor the preselected tubes for hydrogen and helium.

Tables A2-6 and A2-7 list the operating conditions for which the tube vent and purge cart system and the overpack storage tubes have been designed.

Safety Considerations. The overpack storage tubes, bases, and embeds are classified as safety class and important-to-safety Category A. The overpack storage tube plug, lower flanges and the bottom impact absorbers are classified safety significant. All other components of the overpack storage tube assemblies are general service. The floor electrical plugs and helium fill station are general service. The hose, HEPA filter, and interconnecting equipment piping of the tube vent and purge cart are classified safety significant, and the remainder are general service (Figure A2-48).
<table>
<thead>
<tr>
<th>Component</th>
<th>System design conditions</th>
<th>System operating conditions</th>
<th>Design operating environment</th>
<th>Operating environment</th>
<th>Material</th>
<th>Corrosion allowance</th>
</tr>
</thead>
<tbody>
<tr>
<td>HEPA filter (F1-3)</td>
<td>150 lb/in² gauge, 175 °F</td>
<td>5 lb/in² gauge to 0 lb/in² gauge, 110 °F</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>304 stainless steel</td>
<td>None</td>
</tr>
<tr>
<td>Flex hose (SP/1005)</td>
<td>100 lb/in² gauge to 10 torr, 125 °F</td>
<td>5 lb/in² gauge to 0 lb/in² gauge, 110 °F</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>304L stainless steel</td>
<td>None</td>
</tr>
<tr>
<td>Flex hose (SP/1025)</td>
<td>100 lb/in² gauge to 10 torr, 125 °F</td>
<td>5 lb/in² gauge to 0 lb/in² gauge, 110 °F</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>304L stainless steel</td>
<td>None</td>
</tr>
<tr>
<td>H₂ monitor (AE-351)</td>
<td>100 lb/in² gauge to 10 torr, 125 °F</td>
<td>5 lb/in² gauge to 0 lb/in² gauge, 110 °F</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>304-series stainless steel piping</td>
<td>None</td>
</tr>
<tr>
<td>Gas monitor (RGM-622)</td>
<td>100 lb/in² gauge to 10 torr, 125 °F</td>
<td>5 lb/in² gauge to 0 lb/in² gauge, 110 °F</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>304-series stainless steel piping</td>
<td>None</td>
</tr>
<tr>
<td>Oxygen monitor (AA-362)</td>
<td>100 lb/in² gauge to 10 torr, 125 °F</td>
<td>5 lb/in² gauge to 0 lb/in² gauge, 110 °F</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>304-series stainless steel piping</td>
<td>None</td>
</tr>
<tr>
<td>CAM (CAM-596)</td>
<td>100 lb/in² gauge to 10 torr, 125 °F</td>
<td>5 lb/in² gauge to 0 lb/in² gauge, 110 °F</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>304 stainless steel</td>
<td>None</td>
</tr>
<tr>
<td>Heat exchanger (HX-1)</td>
<td>100 lb/in² gauge to 10 torr, 175 °F</td>
<td>5 lb/in² gauge to 0 lb/in² gauge, 110 °F</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>304 stainless steel</td>
<td>None</td>
</tr>
<tr>
<td>Vent piping (MCO-1/2&quot;-TPS-101-SS)</td>
<td>150 lb/in² gauge to 10 torr, -20 °F to 400 °F</td>
<td>5 lb/in² gauge to 0 lb/in² gauge, 110 °F</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>304L stainless steel ASTM A312* S40</td>
<td>None</td>
</tr>
<tr>
<td>Vent piping (MCO-1/2&quot;-TPS-102-SS)</td>
<td>150 lb/in² gauge to 10 torr, -20 °F to 400 °F</td>
<td>5 lb/in² gauge to 0 lb/in² gauge, 110 °F</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>304L stainless steel ASTM A312* S40</td>
<td>None</td>
</tr>
<tr>
<td>Vent piping (MCO-1/2&quot;-TPS-103-SS)</td>
<td>150 lb/in² gauge to 10 torr, -20 °F to 400 °F</td>
<td>5 lb/in² gauge to 0 lb/in² gauge, 110 °F</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>304L stainless steel ASTM A312* S40</td>
<td>None</td>
</tr>
<tr>
<td>Vent piping (HE-3/4&quot;-TPS-110-L)</td>
<td>150 lb/in² gauge, -20 °F to 400 °F</td>
<td>20 lb/in² gauge, 60 °F to 85 °F</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel ASTM A53b S80</td>
<td>0.065 in.</td>
</tr>
<tr>
<td>Vent piping (HE-3/4&quot;-TPS-111-L)</td>
<td>150 lb/in² gauge, -20 °F to 400 °F</td>
<td>20 lb/in² gauge, 60 °F to 85 °F</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel ASTM A53b S80</td>
<td>0.065 in.</td>
</tr>
<tr>
<td>Inert gas piping (HE-3/4&quot;-TPS-113-L)</td>
<td>150 lb/in² gauge, -20 °F to 400 °F</td>
<td>35 lb/in² gauge, 60 °F to 85 °F</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel ASTM A53b S80</td>
<td>0.065 in.</td>
</tr>
</tbody>
</table>

This table is for information only and does not provide operational controls.

Note: The cart has all of the equipment listed in the table.


CAM = continuous air monitor.
HEPA = high-efficiency particulate air (filter).
### Table A2-7. Overpack Storage Tube Design and Operating Conditions.

<table>
<thead>
<tr>
<th>Component</th>
<th>System design conditions</th>
<th>System operating conditions</th>
<th>Design operating environment</th>
<th>Operating environment</th>
<th>Material</th>
</tr>
</thead>
<tbody>
<tr>
<td>Overpack tube</td>
<td>0 to 75 lb/in² gauge 220 °F</td>
<td>0 to 22 lb/in² gauge 115 °F</td>
<td>-27 °F to 200 °F</td>
<td>-27 °F to 115 °F</td>
<td>Carbon steel SA-671</td>
</tr>
<tr>
<td>Tube plug body</td>
<td>5 °F to 115 °F, top 220 °F, bottom</td>
<td>60 °F to 105 °F, top 204 °F, bottom</td>
<td>5 °F to 115 °F</td>
<td>60 °F to 105 °F</td>
<td>Carbon steel, concrete ASME Section III, Division 1a</td>
</tr>
<tr>
<td>Tube plug seals</td>
<td>-65 °F to 300 °F 1 x 10³ rad/yr gamma</td>
<td>60 °F to 105 °F 4.83 x 10³ rad/yr gamma</td>
<td>-20 °F to 300 °F 6 lb/in² gauge to 10 torr</td>
<td>150 °F, 5 lb/in² gauge</td>
<td>Ethylene propylene</td>
</tr>
<tr>
<td>Tube plug piping</td>
<td>300 °F, 150 lb/in² gauge</td>
<td>175 °F, 5 lb/in² gauge (normal) 300 °F, 6 lb/in² gauge (peak)</td>
<td>5 °F to 115 °F</td>
<td>60 °F to 105 °F</td>
<td>Carbon steel ASTM A106b</td>
</tr>
<tr>
<td>Tube plug pressure gauge</td>
<td>200 °F, 0 to 15 lb/in² gauge</td>
<td>175 °F, 5 lb/in² gauge (normal) 300 °F, 6 lb/in² gauge (peak)</td>
<td>5 °F to 115 °F</td>
<td>60 °F to 105 °F</td>
<td>316 stainless steel</td>
</tr>
<tr>
<td>Bottom impact absorber</td>
<td>330 °F, top 330 °F, bottom 200 R/h 1.3 x 10⁷ Rads</td>
<td>220 °F (average)</td>
<td>220 °F (average)</td>
<td>30 °F to 115 °F</td>
<td>Manufacturer standard</td>
</tr>
<tr>
<td>Overpack storage tube base</td>
<td>220 °F</td>
<td>115 °F</td>
<td>-27 °F to 200 °F</td>
<td>60 °F to 115 °F</td>
<td>Carbon steel</td>
</tr>
<tr>
<td>Overpack storage tube embed</td>
<td>220 °F</td>
<td>115 °F</td>
<td>-27 °F to 200 °F</td>
<td>60 °F to 115 °F</td>
<td>Carbon steel</td>
</tr>
</tbody>
</table>

This table is for information only and does not provide operational controls.


A2.6 CONFINEMENT SYSTEMS

This section defines the confinement requirements, and identifies and describes the SSCs that perform confinement functions for the SNF stored in MCOs in the CSB. These confinement systems for sealed MCOs containing SNF meet the requirements of DOE Order 6430.1A, Sections 1300-7 “General Requirements: Confinement Systems,” and 1320-5 “Irradiated Fissile Material Storage Facilities: Confinement Systems.” A confinement system is “the barrier and its associated systems (including ventilation) between areas containing hazardous materials and the environment or other areas in the facility that are normally expected to have levels of hazardous materials lower than allowable concentration limits.” The following is a discussion of these confinement requirements.

A2.6.1 Requirements

DOE orders provide requirements and recommendations for the appropriate level of confinement needed for fuel storage. DOE Order 6430.1A, Section 1300-7, “General Requirements: Confinement Systems,” and 1320-5, “Irradiated Fissile Material Storage Facilities: Confinement Systems,” address the confinement appropriate for SNF fuel storage at the CSB. Section 1300-7 states:

“Confinement systems shall accomplish the following:

- Minimize the spread of radioactive and other hazardous materials within the unoccupied process areas
- Prevent, if possible, or else minimize the spread of radioactive and other hazardous materials to occupied areas
- Minimize the release of radioactive and other hazardous materials in facility effluents during normal operation and anticipated operational occurrences
- Limit the release of radioactive and other hazardous materials resulting from design basis accidents including severe natural phenomena and man-made events in compliance with the guidelines contained in Section 1300-1.4.2, Accidental Releases.”

Section 1300-7 are provides other general requirements.

- The HVAC system shall maintain a controlled continuous airflow pattern from the environment into the confinement building, from uncontaminated areas to potentially contaminated areas, and then to contaminated areas.
The number and arrangement of confinement barriers are determined on a case-by-case basis with the final objective being a practical design that achieves the confinement objectives.

The confinement systems shall perform their required functions for the design basis accidents they are required to withstand.

At least one confinement system remains fully functional following any credible design basis accident.

Section 1320-5 states: “The following provisions are typical for an Irradiated Fissile Material Storage Facilities (IFMSF) confinement system. The actual confinement system requirements for a specific IFMSF shall be determined on a case-by-case basis. In general, the primary confinement shall be the IFMSF cladding or canning. Secondary confinement shall be established by the facility buildings that enclose the dry storage area and/or the storage pool and auxiliary systems.”

Other general requirements from Section 1320-5 include the following.

- Special design features shall be considered for safe handling of irradiated fissile material to protect against dropping of shipping casks.
- Penetrations of the secondary confinement barrier shall have positive seals to prevent the migration of contamination.
- Ventilation systems shall provide for inlet air filtration, and the exhaust air shall have HEPA filters.

In summary, Section 1320-5 identifies the SSCs that accomplish the confinement objectives. In general the confinement barriers are identified on a case-by-case basis.

A2.6.2 Safety Classification of Confinement Systems

The hazard and accident analysis performed in Chapter A3.0 determines the safety classification for all SSCs in the CSB, including the confinement systems. The safety classification process uses guidance from HNF-PRO-704, Hazard and Accident Analysis Process, and applies the graded approach methodology of DOE-STD-3009-94, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports. Classification of a confinement system as safety class, safety significant, or general service depends on the SSC’s preventive, mitigative, or worker safety functions for reducing adverse effects on the environment, the public, onsite workers, and facility workers. As shown in Chapter A3.0, there are no postulated releases with consequences exceeding offsite release limits. Therefore, no SSCs are required to have a safety-class confinement function. However, several SSCs have been
determined to have other safety-class functions and thus are classified as safety class. The one exception is the MCO sample station hood as described in Section A2.5.1.4.

A2.6.3 Confinement During Canister Storage Building Operations

The DOE Order 6430.1A requirements identified in Section A2.6.1 provide a flexible structure for determining the confinement requirements based upon the potential hazards and the propensity for radiological release. The SNF stored within the MCO does not have intact cladding so that no credit may be taken for the cladding as a confinement barrier. Therefore, the MCO provides the primary confinement barrier. The secondary confinement barrier during transportation cask–MCO handling in the trailer vestibule and the MCO service station is the transportation cask (Figure A2-49). For other operations within the CSB (Figure A2-50), the equipment encasing the MCO provides the secondary confinement barrier (Figure A2-49) as listed in Table A2-8. Finally, the building structure and ventilation system provides additional confinement, defense in depth, for the SNF. As explained in greater detail in the following sections, the confinement barriers for the CSB accomplish the confinement objectives required by DOE Order 6430.1A.

Operations involving MCOs containing SNF are conducted within the CSB operating shelter in five primary areas: the trailer vestibule, the MCO service station, the MHM, the sampling/weld stations, and the storage tubes. The design of the CSB provides primary and secondary confinement barriers during all normal operations. Figure A2-49 depicts the confinement barriers during normal CSB operations (Figure A2-50). The primary and secondary confinement barrier systems are listed in Table A2-8. These include receipt, handling, and interim storage of the MCOs containing SNF. The confinement systems in Table 2-8 include passive barriers, active barriers, and confinement protection features used by operations to assure that the confinement systems are functional.

A2.6.3.1 Multi-Canister Overpack Confinement Features. The MCO is a mechanically sealed pressure vessel before it is shipped to the CSB. The MCO is closed with a mechanical locking ring at the K Basins before the MCO is shipped to the CVDF. The CVDF processes the MCOs through process ports in the shield plug using a cold vacuum drying process. All four ports of the MCO are mechanically sealed with blind gasketed cover plates at the CVDF. This makes the process valves and pressure relief features of the MCO inactive while the MCO is being transported and during all phases of operation at the CSB. After a cap is welded to the MCO, the sealed welded MCO (cap and shell) provides the permanent primary confinement barrier for the SNF.

During all phases of normal CSB operations the MCO is credited with providing the primary confinement barrier against accidental releases of SNF or particulate. This barrier is designed to remain essentially leak tight during all normal and credible events except drops or shears from cranes and other handling equipment, as discussed in SNF-4042, Evaluation of Accident Frequencies at the Canister Storage Building. The leak rates for which the MCO is
<table>
<thead>
<tr>
<th>Operations</th>
<th>Trailer vestibule</th>
<th>MCO service station</th>
<th>MHM</th>
<th>Storage tubes</th>
<th>Sampling/weld station</th>
</tr>
</thead>
<tbody>
<tr>
<td>Passive barriers</td>
<td>Primary</td>
<td>MCO, mechanical seal</td>
<td>MCO, mechanical seal</td>
<td>MCO, mechanical seal or welded cap</td>
<td>MCO</td>
</tr>
<tr>
<td></td>
<td>Secondary</td>
<td>Cask</td>
<td>MHM cask cavity</td>
<td>Storage tube, tube plug, breather filter</td>
<td>Sample hood (HEPA), sampling/weld pit, shield plate</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Cask receiving pit, shield hatch and MCO guide assembly, mobile service station test (off-normal use only)</td>
<td>Storage tube, tube plug, breather filter</td>
<td>Overpack storage tube, tube plug, breather filter, lockdown device</td>
<td></td>
</tr>
<tr>
<td>Additional</td>
<td>Operating area shelter</td>
<td>Operating area shelter</td>
<td>Operating area shelter</td>
<td>Operating area shelter</td>
<td>Operating area shelter</td>
</tr>
<tr>
<td>Active barriers</td>
<td>HVAC (HEPA)</td>
<td>HVAC (HEPA)</td>
<td>MHM vent system HVAC (HEPA)</td>
<td>HVAC (HEPA)</td>
<td>Sampling vent system, pressure test of sample cart, HVAC (HEPA)</td>
</tr>
<tr>
<td>Confinement protection features</td>
<td>Cask service system pressure check</td>
<td>MHM interlocks</td>
<td>MHM interlocks, surveillance, weld leak test (future)</td>
<td>MHM interlocks, surveillance</td>
<td>Plug seal test, tube vent and purge cart pressure test</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>Process valve operator, hood flow indicator</td>
</tr>
</tbody>
</table>

HEPA = high-efficiency particulate air (filter).
HVAC = heating, ventilation, and air conditioning.
MCO = multi-canister overpack.
MHM = multi-canister overpack handling machine.
designed are $\leq 1.0 \times 10^{-3}$ cm$^3$/s in the mechanically sealed configuration, and $\leq 1.0 \times 10^{-7}$ cm$^3$/s after welding the canister cover assembly.

A2.6.3.2 Confinement During Trailer Vestibule Operations. Trailer vestibule operations include off-loading the transportation cask/MCO from the transporter, and transferring the transportation cask–MCO from the trailer vestibule to the cask receiving pit. These operations are described in Section A2.5.1. For operations in the trailer vestibule area, the MCO shell provides the primary confinement barrier with the transportation cask providing secondary confinement functions.

A2.6.3.3 Confinement During Multi-Canister Overpack Service Station Operations. Operations in the MCO service station are related with preparing the transportation cask for MCO removal by the MHM. These operations include checking the internal pressure of the transportation cask on the transportation trailer or in the cask receiving pit before removing the cask lid. During operations in the cask receiving pit, primary confinement is provided by the MCO shell with the cask receiving pit and the shield hatch and MCO guide assembly, providing secondary confinement functions during preparation of the MCO for removal by the MHM.

Venting casks having "a higher than normal pressure" are considered for recovery operations from abnormal conditions. In addition to the confinement provided by the MCO shell and the MCO service station equipment (pit, shield hatch and MCO guide assembly, and tent for off-normal use), the cask servicing system provides secondary confinement functions during cask servicing operations.

A2.6.3.4 Confinement During Multi-Canister Overpack Handling Machine Operations. Once the transportation cask has been prepared for MCO removal by the MHM, the MHM's tube plug grapple is used to remove and replace the shield hatch and MCO guide assembly's center shield hatch plate (see Figure A2-17). The MCO hoist and grapple is used to withdraw the MCO from the transportation cask. The MHM can be used to transport the MCO to the sample station, weld station, or to a standard storage tube for staging. These travel paths are depicted on Figure A2-50.

During MCO transfers between the MHM and a storage tube, air from the operating area and from the storage tube is drawn through the MHM cask space and into the MHM's HEPA-filtered extract system. Control of potential incidental contamination from an MCO is maintained by the MHM's onboard, HEPA-filtered, extract system during MCO handling. Air from the MHM exhaust system discharges into the CSB operating area and is exhausted through the CSB HVAC system (Figure A2-51), which also contains HEPA filters. The MCO shell provides the primary confinement barrier with the MHM cask cavity and MHM extract system providing secondary confinement. The operating area shelter and the HVAC system provide additional confinement functions.

A2.6.3.5 Confinement During Sampling/Weld Station Operations. MCOs are transported by the MHM to a sampling/weld station where monitoring of MCO temperature and internal
pressure, and gas sampling occurs. After lowering an MCO into the sampling/weld station, a
sampling/weld station gantry crane removes the center shield plate and shield halves. A sample
hood is positioned over the MCO and sample lines attached to MCO process port 2. This MCO
process port connects to the MCO’s internal HEPA filter. A HEPA filtered sample hood exhaust
system provides secondary confinement to the MCO and sample lines. A sample cart connected
to the sample hood process HEPA filter provides the equipment for taking samples and
diluting/exhausting the internal MCO gases. The MCO and MCO internal HEPA filter provide
primary confinement for the SNF. The flexible line up to the HEPA filter, the sample hood HEPA
filter, the sample cart, sampling/weld pit, and sampling vent system provide secondary
confinement functions. The operating shelter and HVAC system also provide confinement
functions. For a short period of time between removal of the sampling/weld station shield plate
and installation of the sample hood, the secondary confinement function is provided by the
operating shelter and HVAC system.

A2.6.3.6 Confinement During Standard Storage Tube Operations. The MCO shell provides
primary confinement for SNF during storage in the standard storage tubes with the storage tube
and tube plug providing secondary confinement functions. An MCO that is to be monitored is
removed from its storage tube and transported by the MHM to a sampling/weld station for
sampling activities. The standard storage tubes are maintained at atmospheric pressure through a
filtered port in the tube plug, providing filtered inflow and outflow of air mixing with the facility’s
HEPA-filtered HVAC system. The tube plug design includes a tube plug filtered exhaust port, a
tube plug seal, and a valved port. The operating shelter and HVAC system also provide
confinement functions.

A2.6.4 Confinement During Storage Tube Surveillance

Operations described in Section A2.5 require long-term surveillance for the standard
storage tubes. These surveillance activities may include periodic radiation surveying of standard
tube-filtered exhaust ports and occasional monitoring of standard tube atmosphere using the tube
vent and purge cart. As mentioned in the previous section, the MCO shell provides the primary
confinement barrier for the SNF with the standard storage tube and standard tube plug providing
secondary confinement. The operating shelter and the HVAC system provide additional
confinement functions.

The operating shelter provides additional confinement for radioactive particulate and,
together with the shield wall and perimeter curb, provides additional shielding for radiation
control. The operating area shelter is a weatherproof structure having walls and a roofing system
that are resistant to wind-driven missiles and tornado wind loads. The operating shelter provides
additional confinement for radioactive particulate and, together with the shield wall and perimeter
curb, provides additional shielding for radiation control. However, it is not relied upon as a
mitigating feature against particulate releases caused by the postulated accident sequences.
In addition, the building ventilation system contains HEPA filters for removal of airborne
particulate. The operating area ventilation system filters would protect the public, environment,
and onsite workers by filtering radioactive particulate should a release occur from any of the
confinement systems. Therefore, the operating area shelter and the operating shelter HVAC
system provide additional confinement features for confinement of SNF and SNF particulate.

A2.6.5 Confinement During Overpack Storage Tube Operations

Overpack storage tube operations only occur when CSB operations detect an abnormal or
accident condition for an MCO. This may occur when the cask servicing system detects an
abnormal MCO. Recovery actions are determined, and the MCO subsequently is moved into an
overpack storage tube for monitored storage. The overpack storage tube operations include
checking the internal pressure of the overpack storage tube, checking the overpack tube plug
seals, monitoring the tube atmosphere for hydrogen concentration, purging the overpack storage
tube as needed to lower the hydrogen concentration, and adding inert gas. Venting, monitoring,
and reinerting the overpack storage tube is required when periodic checks indicate an abnormal
tube pressure. These actions also are required for an MCO that has vented hydrogen into the
overpack storage tube.

A2.7 SAFETY SUPPORT SYSTEMS

The principal safety support systems are identified and described in this section. The
descriptions include the purpose, an overview, principal components, operations, and control
functions.

A2.7.1 Fire Protection Features

The CSB structural system, roof, and exterior wall system are made of noncombustible
materials. This gives the building a construction type of Uniform Building Code (ICBO 1994),
Type II-N (nonrated). In any interior areas that have finishes, the finishes have a flame spread
index less than 25 when tested in accordance with ASTM E84, Test for Surface Burning
Analysis for the Canister Storage Building, by the Authority Having Jurisdiction (DOE, Richland
Operations Office) determined that an H-7 occupancy classification was more appropriate.
A deviation request, Letter 9655218, Project W-379, Spent Nuclear Fuel Canister Storage
Request for Deviation from the United States Department of Energy Order 6430.1A —
Automatic Sprinkler Protection Requirements (Williams 1996), was submitted to DOE, Richland
Operations Office, to allow the application of NFPA 101, Safety to Life from Fire in Buildings
and Structures, and NFPA 220, Standards on Types of Building Construction, in lieu of the
Uniform Building Code (ICBO 1994) requirements for definitions, occupancy classification, fire
ratings, and types of construction. Fluor Daniel Hanford, Incorporated has received approval of
the deviation request (Sellers 1996). The roof structural and steel column members in the truck
vestibule area and the two steel column members within the operations area (adjacent to the

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Megadoor™) have a protective coating (minimum 3/4-in. thickness of spray-on coating on metal lath) providing a one-hour fire rating. The one-hour fire protective coating protects against the potential diesel spill fire that could cause structural failure of the roof supports and the steel column support members.

The administrative controls summarized below are based on analysis contained in the fire hazards analysis (HNF-SD-SNF-FHA-002).

- Combustible materials inherent to all equipment proposed for use within the operating area shall be ALARA. This includes limiting the use of materials with combustible components, and whenever possible, replacing them with either noncombustible or fire resistant or retardant components.

- Damage potential resulting from a vehicle fuel spill and resulting fire in the trailer vestibule shall be limited by restricting the receiving crane by administrative procedure or control from that area during those periods when a cask transportation truck and tractor is staged (located) within the trailer vestibule. A 3-in.-high dam in the receiving crane rail trenches prevents diesel flow into the load-in/load-out and operating areas.

- Transient combustible materials shall be limited to the minimum type and quantities required to perform the transient activity or operation and shall be removed upon completion of that activity or operation.

- Potential fire size resulting from a transient combustible fire shall be controlled by restricting the size and quantity of potentially combustible materials (i.e., MCOs wrapped in fire-retardant polyethylene or trash bags filled during normal operations) and maintaining the minimum separation distances identified in HNF-SD-SNF-FHA-002.

- The uncontrolled leakage of fluid (i.e., spray) from the receiving crane or MHM shall require the immediate safe shutdown of operations until corrective action to seal the leak is completed and the fire hazard inherent to the leak is eliminated.

The fire protection system for the CSB is designed to protect the support building from damage. A very early smoke detection alarm system draws air from the upper portion of the operating area through plastic piping and detects smoke. The fire alarm control panel is located in the entry hallway outside of the facility manager's office (room 28). The fire alarm features include transmission of signals to the Hanford Fire Department via a radio fire alarm reporter; annunciation of separate and distinct fire, supervisory, and trouble alarms; annunciation of local building fire alarms; and shutdown of appropriate HVAC units initiated upon receipt of any alarm.

The standard response to an alarm condition in the 200 East Area will be by the Hanford Fire Department from the 200 Area Fire Station. Response time to the CSB will be...
approximately 5 minutes. When manned, a crew from the 100 Area Fire Station will be
dispatched simultaneously with an estimated response time of 10 minutes. Vehicle access to the
facility will be provided by a paved access road that is included as part of the facility construction.
The Hanford Fire Department is fully staffed, trained, and equipped.

A pre-fire plan developed by the Hanford Fire Department will be issued before operation
of the facility. This plan will include special purpose systems for the operating area as the first
approach to fire suppression because of the restriction of using large quantities of water. Should
these special purpose systems fail to extinguish a fire, water hose streams may be used as a last
resort. Chapter A6.0 establishes the criticality safety fire fighting category and discusses any
restrictions for using water to extinguish a fire involving fissionable material.

The common wall section between the operations area and the support area building is
designed to provide a 2-hour Underwriters Laboratories-rated fire protection. One and one-half
hour rated fire dampers are provided at the exhaust and return registers of the operating area
(FD-1 and FD-2) and also at the supply to the utility rooms (FD-3 and FD-4). The fire control
panel is powered from the uninterruptible power supply (UPS).

The rooms of the CSB support area building are protected against fire damage by the fire
walls and the automatic, wet-pipe sprinkler system. The wet-pipe sprinkler system has been
designed to meet the requirements of DOE Order 6430.1A, Section 1530-3.1, and NFPA 13,
Installation of Sprinkler Systems. The design flow rate is 0.19 gal/min-sprinkler head (2,500 ft²),
ordinary hazard, Group 3. The fire alarm system may be activated by a manual alarm box.
Provisions have been included for future activation mechanisms as needed. The two utility rooms
are separated from the support area building and each other by a 2-hour rated fire wall. A space
heater in each utility room provides freeze protection for the sprinkler system. A 6-in. fire water
supply system begins at the north side of the CSB support area building using a connection to a
10-in. fire water main (Figure A2-4). DOE Order 6430.1A, Section 1530-99.0, requires a
minimum of two reliable, independent sources of water. The second source is provided by a
44,000-gal storage tank and pumping system dedicated to the fire water loop around the CSB and
Building 2704-HV. A post-indicating valve provides isolation of the 6-in. CSB fire water line
from the fire water main if required. Bollards, located adjacent to the post-indicating main shut-
off valve, protect the valve from accidental collision with off-road vehicle traffic. Other bollards
near the main shut-off valve protect the nearby cathodic test stations. A cathodic protection
system has been installed on the underground lines to prevent corrosion of the carbon steel water
lines (potable and sanitary). Other water mains include the 10-in. and 12-in. fire and raw water
lines to the north and east of the facility and smaller underground lines in the immediate area
around the facility. The routing of the sprinkler header is along the support area building corridor
ceiling. Runoff from sprinklers in uncontaminated areas goes to the outside of the building.
Runoff from fire sprinkler actuation in potentially contaminated areas (regulated change rooms) is
captured in the support area building in recessed floor sumps. Sleeves, pipes in a sleeve, and
conducts in the support area building pits are sealed to prevent runoff from leaving the sprinkled
areas. Water released by a rupture or actuation of the wet-pipe sprinkler fire system will be
collected in recessed floor sumps in the support area building (control, step-off pad, count,
monitoring and decontamination, and regulated change rooms).
Additional fire protection features include fire extinguishers located in the support area (electrical equipment, filter room, both air locks, and corridor), the trailer vestibule area, and the loading and staging area (Figure A2-3) that are mounted in accordance with NFPA 10. Also, manual pull boxes are mounted in accordance with NFPA 10 in the control room, at the east support area entrance, at the support area maintenance area, and the northeast operating area exit, and two are mounted in the sampling/weld station area (Figure A2-3).

Range fires do not present an exposure potential to the CSB. A clear space of at least 60 ft is provided on all sides of the CSB (HNF-SD-SNF-FHA-002). Unpaved areas around the CSB are overlaid with crushed rock surface.

Water leaking from the underground water supply lines could cause water accumulation, drainage problems, soil erosion, and CSB building foundation failure if the leak is not detected and the leaking line is not isolated from the water main. Periodically checking the soil level surrounding the facility and checking the soil level after an earthquake provide for early detection of water leaks and prevent significant soil erosion.

Table A2-9 lists the operating conditions for which the fire protection system has been designed.

**Safety Considerations.** The fire protection system is classified general service.

### A2.7.2 Heating, Ventilation, and Air Conditioning System

The HVAC system is designed to provide, along with physical barriers, part of the CSB contamination confinement system and contamination control within the CSB. The HVAC system provides a controlled pressure gradient flow of air from outside the CSB inward through uncontaminated areas to potentially contaminated areas of the building and out through HEPA filters and a monitored exhaust. The HVAC system also provides climate control to ensure that environmental conditions in the CSB are maintained in the required ranges for personnel and equipment. The HEPA-filtered service station enclosure discharges air into the operations area. The operations area air exchange rate is approximately 1.5 volume changes per hour.

The HVAC composite flow diagram can be seen on Figure A2-51. The HVAC system has its own programmable controller. The HVAC system is manually adjusted to achieve the operating area negative pressure. Dampers automatically compensate for changes in filter pressure drop or other sources of exhaust from the CSB. The setpoint for starting the second exhaust fan, in the event of failure of the lead fan, is 50% of normal flow or 4,500 ft³/min. The HVAC system comprises two subsystems, the operating area HVAC subsystem and the support area HVAC subsystem.
<table>
<thead>
<tr>
<th>Component</th>
<th>System design conditions</th>
<th>System operating conditions</th>
<th>Design operating conditions</th>
<th>Operating environment</th>
<th>Material</th>
<th>Corrosion allowance</th>
</tr>
</thead>
<tbody>
<tr>
<td>Piping</td>
<td>100 °F, 150 lb/in² gauge</td>
<td>60 °F to 85 °F, 85 to 105 lb/in² gauge</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td>Valves</td>
<td>-20 °F to 150 °F, 175 lb/in² gauge</td>
<td>60 °F to 85 °F, 85 to 105 lb/in² gauge</td>
<td>32 °F to 130 °F</td>
<td>60 °F to 85 °F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
</tbody>
</table>

This table is for information only and does not provide operational controls.

NA = not applicable.
A2.7.2.1 Operating Area Heating, Ventilation, and Air Conditioning. The operating area HVAC subsystem provides ventilation and air conditioning for the CSB operating area and the HVAC equipment room. The operating area HVAC system is designed to maintain a negative pressure with regard to the outside environment to ensure airflow from areas of lower contamination potential to areas of higher contamination potential. The operating area HVAC system provides approximately 44,000 ft³/min of supply air and recirculates approximately 35,500 ft³/min of air through the operating area and HVAC equipment rooms. The HVAC system has continuous outside air makeup of 6,750 ft³/min and a HEPA-filtered exhaust of 8,970 ft³/min to 9,000 ft³/min that is continuously monitored for contamination. Outside air is drawn through roof-mounted intakes and an electric duct heater to air handling units where it is mixed with air recirculated from within the building. The exhaust air fans are equipped with backdraft dampers (BD-1 and BD-2) that are counterweighted to allow flow in only one direction to prevent backflow through the fan when it is not operating, damage to the fan, or possible spread of contamination.

During cold weather conditions, electric heating elements (present in the electric duct heater and in each operating area HVAC air handling unit) operate in a controlled, cascading fashion to maintain the operating area air temperature within the normal range above the low-temperature setpoint. For warm weather conditions, external, air-cooled condenser/compressor units that supply refrigerant to cooling vanes in the air handlers maintain the operating area air temperature in the normal range below the high-temperature setpoint. Inlet air coolers CU-001 and CU-002 are programmed using the LP-HV-001 HVAC panel board with one cooler designated as lead and the other as lag condensing unit. The controls turn on the first compressor of the lead condensing unit. A second compressor of the lead unit is started if the compressor is not providing adequate cooling after a set time delay. The lag condensing unit automatically starts if the lead condensing unit cannot satisfy adequate cooling after a given time delay.

Table A2-10 lists the operating conditions for which the operating area HVAC has been designed.

A2.7.2.2 Support Area Heating, Ventilation, and Air Conditioning. After passing through the operating area or the HVAC equipment room, air is exhausted from the support area HVAC through HEPA filters and an exhaust fan and out the main exhaust stack where continuous radiation monitoring records contamination, if any, in the air. Heater blankets in each exhaust HEPA filter housing prevent the accumulation of condensation on the filters. Ember screens upstream of the HEPA filters protect the filters from potential fire damage and meet the requirements of the *DOE Filter Plenum Fire Protection Standard* (DOE 1993).

The operating area HVAC airflow is balanced by setting manual airflow dampers in the operating area intake, recirculation, and HVAC equipment room ducts to achieve the desired differential pressure with a constant exhaust fan airflow. Manual damper MVD-1 maintains the balance of intake air to recirculating airflow for the operating area HVAC subsystem; manual damper MVD-2 maintains the balance of flows exiting the HVAC equipment room between that entering the exhaust flowpath and that recirculating to the operating area HVAC subsystem;
Table A2-10. Operating Area Heating, Ventilation, and Air Conditioning Design and Operating Conditions.

<table>
<thead>
<tr>
<th>Component</th>
<th>System design conditions</th>
<th>System operating conditions</th>
<th>Design operating environment</th>
<th>Operating environment</th>
<th>Material</th>
<th>Corrosion allowance</th>
</tr>
</thead>
<tbody>
<tr>
<td>Air handling unit (CSB-AH-001)</td>
<td>4 °F to 101 °F</td>
<td>60 °F to 85 °F</td>
<td>60 °F to 85 °F</td>
<td>55 °F to 95 °F</td>
<td>Galvanized steel 18 gauge ASTM A653*</td>
<td>None</td>
</tr>
<tr>
<td>Air handling unit (CSB-AH-002)</td>
<td>4 °F to 101 °F</td>
<td>60 °F to 85 °F</td>
<td>60 °F to 85 °F</td>
<td>55 °F to 95 °F</td>
<td>Galvanized steel 18 gauge ASTM A653*</td>
<td>None</td>
</tr>
<tr>
<td>HEPA filter housing (CSB-PF-001)</td>
<td>55 °F to 95 °F</td>
<td>60 °F to 85 °F</td>
<td>60 °F to 85 °F</td>
<td>55 °F to 95 °F</td>
<td>304 stainless steel</td>
<td>None</td>
</tr>
<tr>
<td>HEPA filter housing (CSB-PF-002)</td>
<td>55 °F to 95 °F</td>
<td>60 °F to 85 °F</td>
<td>60 °F to 85 °F</td>
<td>55 °F to 95 °F</td>
<td>304 stainless steel</td>
<td>None</td>
</tr>
<tr>
<td>Exhaust fan (CSB-EF-001)</td>
<td>55 °F to 95 °F</td>
<td>55 °F to 95 °F</td>
<td>60 °F to 85 °F</td>
<td>55 °F to 95 °F</td>
<td>Carbon steel</td>
<td>None</td>
</tr>
<tr>
<td>Exhaust fan (CSB-EF-002)</td>
<td>55 °F to 95 °F</td>
<td>55 °F to 95 °F</td>
<td>60 °F to 85 °F</td>
<td>55 °F to 95 °F</td>
<td>Carbon steel</td>
<td>None</td>
</tr>
<tr>
<td>Duct heater (CSB-EH-002)</td>
<td>4 °F to 101 °F</td>
<td>-27 °F to 60 °F</td>
<td>60 °F to 85 °F</td>
<td>55 °F to 95 °F</td>
<td>Carbon steel</td>
<td>None</td>
</tr>
<tr>
<td>Supply and return duct work</td>
<td>4 °F to 101 °F</td>
<td>55 °F to 95 °F</td>
<td>60 °F to 85 °F</td>
<td>55 °F to 95 °F</td>
<td>Galvanized steel ASTM A653*</td>
<td>None</td>
</tr>
<tr>
<td>Exhaust duct work</td>
<td>55 °F to 95 °F</td>
<td>55 °F to 95 °F</td>
<td>60 °F to 85 °F</td>
<td>55 °F to 95 °F</td>
<td>304 stainless steel</td>
<td>None</td>
</tr>
</tbody>
</table>

This table is for information only and does not provide operational controls.


HEPA = high-efficiency particulate air (filter).
manual damper MVD-3 sets the airflow entering the HVAC equipment room; and manual damper MVD-4 maintains the airflow to the "entry-related" rooms of the support area HVAC subsystem. The operating area HVAC controls the positions of the inlet vane damper of the operating area exhaust fan, varying the overall HVAC exhaust to maintain the desired constant exhaust fan discharge airflow. If both exhaust fans are shut down, the operating area HVAC air handlers shut down automatically (loss-of-flow controller FIC-281 50% exhaust flow permissive); if the operating exhaust fan is shut down, the other ("lag") exhaust fan automatically starts, followed by a restart of the air handlers when the 50% exhaust flow permissive is again met on loss-of-flow controller FIC-281. By shutting down the air handling units and allowing only an exhaust fan to operate in a low exhaust flow condition, an overpressurization of the operating area HVAC system and possible contamination spread is averted.

Automatic, fusible-link fire dampers and duct-mounted smoke detectors provide for worker safety and help prevent the fire from spreading to other areas of the facility. Fire dampers FD-1 and FD-2 are placed to automatically isolate the operating area from the rest of the operating area HVAC. An ember screen attached to FD-2 prevents airborne embers from entering the exhaust air plenum (Figure A2-51). Fire dampers FD-3 and FD-7 (room 018) and FD-4 and FD-8 (room 034) are placed to automatically isolate the utility rooms (room 1 and room 2, respectively) from the support area HVAC by blocking the room HVAC inlet flow ducting and return air dampers. Operation of the fire dampers in the operating area HVAC causes a loss of exhaust flow, with subsequent trip of the air handlers from the 50% exhaust flow permissive setpoint. The smoke detectors are mounted in the operating area HVAC return line and in the support area HVAC recirculation line. A trip of the operating area HVAC duct-mounted smoke detector shuts down the HVAC inlet air handlers (AH-001, AH-002, and AH-003); sounds an alarm within the facility; and transmits a fire alarm by radio from the fire control panel to the central fire alarm system, causing the fire department to respond. An exhaust fan (EF-001 or EF-002) continues to operate and maintains a negative pressure in the CSB. A trip of the support area HVAC duct-mounted smoke detector also performs these actions and sends a signal to the fire department. The fusible-link sprinklers in the support area release a water spray into the support area rooms when the air temperature melts the fusible link. Water flowing through the wet-pipe sprinkling system activates a flow switch in the sprinkler system and transmits a signal to the fire department indicating that the sprinklers have activated.

The support area HVAC subsystem uses forced-air ventilation to maintain desired support area temperatures and required contamination confinement for potentially contaminated rooms (also known as "entry-related rooms" [i.e., those rooms located between airlocks]) in the support area. The support area subsystem serves the control room, corridor, telephone and equipment room, effluent monitor room, utility rooms, electrical equipment room, regulated change room, HPT office, step-off pad and bag room, and the monitoring and decontamination room. Air is drawn from outside the building through roof-mounted intakes and an electric duct heater to an air-handling unit and passes through the support area rooms. Air from entry-related rooms is exhausted through a prefilter and another air-handling unit (AH-004) at a constant 920 ft³/min (nominal) flow rate and is discharged to the HVAC equipment room (where it joins the operating
area HVAC exhaust flow). Air supplied to the support area rooms shown in Figure A2-51 is recirculated.

The support area HVAC subsystem airflow is balanced by setting the manual airflow damper in the duct to the entry-related rooms. The relief damper (which relieves excess air pressure from the support area to the outside) is set to achieve the desired negative differential pressure balanced with the rest of the support area HVAC subsystem in the entry-related rooms. The relief damper relieves air only from the clean air system. Consequently, no air monitoring of air discharged from the relief damper is required. The desired airflow throughout the rest of the support area HVAC subsystem is balanced with a constant exhaust airflow (to the operating area HVAC subsystem). The support area HVAC maintains the desired differential pressures and flows by controlling the speed of the exhaust fan to maintain a constant discharge airflow. Support area HVAC air handler AH-003 will not start until support area HVAC exhaust fan AH-004 flow rate reaches the minimum 50% airflow. If exhaust fan AH-004 is shut down and low flow is sensed, the low-flow interlock automatically shuts down support area HVAC supply air handler AH-003 to prevent overpressurization. If the supply air handler is shut down, the exhaust fan continues to operate at a constant exhaust flow.

An air flow of 4,500 standard ft³/min is induced through the sampling/weld area stations by the operating area HVAC system. This airflow is exhausted through the sampling/weld area ducting to the operating area ventilation system after being HEPA-filtered and monitored.

The HVAC requirements in ANSI/ASME N509 and ANSI/ASME N510 were used in designing the CSB HVAC general-service and safety-significant systems. These design features provide onsite radiological doses that implement the principles of ALARA. Therefore, the NRC equivalency requirements, Item 12, for HVAC systems are met.

Table A2-11 lists the operating conditions for which the support area HVAC has been designed.

A2.7.2.3 Heating, Ventilation, and Air Conditioning Exhaust Stack. The HVAC exhaust stack is 75 ft tall; 2 ft, 4 in. in diameter; and is constructed of carbon steel plate and carbon steel pipe. The stack is formed from tubular sections of steel pipe joined with bolted flanges at two places along its height. The stack wall is a minimum of 0.375 in. thick. The uppermost 22-ft, 6-in. section has steel wind deflector vanes mounted around the outside of the stack. The stack has six capped test ports, 2 in. in diameter, used for airflow rate tests and air sample tests (Figure A2-52).

Safety Considerations. The HVAC system is classified general service.
### Table A2-11. Support Area Building Heating, Ventilation, and Air Conditioning Design and Operating Conditions.

<table>
<thead>
<tr>
<th>Component</th>
<th>System design conditions</th>
<th>System operating conditions</th>
<th>Design operating environment</th>
<th>Operating environment</th>
<th>Material</th>
<th>Corrosion allowance</th>
</tr>
</thead>
<tbody>
<tr>
<td>Air handling unit (CSB-AH-003)</td>
<td>4 °F to 101 °F</td>
<td>60 °F to 80 °F</td>
<td>NA</td>
<td>55 °F to 95 °F</td>
<td>Galvanized steel 18 gauge ASTM A653*</td>
<td>None</td>
</tr>
<tr>
<td>Air handling unit (CSB-AH-004)</td>
<td>55 °F to 95 °F</td>
<td>72 °F to 78 °F</td>
<td>55 °F to 95 °F</td>
<td>55 °F to 95 °F</td>
<td>Galvanized steel 20 gauge AST 653.* 304 stainless steel filter section</td>
<td>None</td>
</tr>
<tr>
<td>Duct heater (CSB-EH-001)</td>
<td>4 °F to 101 °F</td>
<td>-27 °F to 60 °F</td>
<td>60 °F to 104 °F</td>
<td>55 °F to 95 °F</td>
<td>Carbon steel</td>
<td>None</td>
</tr>
<tr>
<td>Distribution duct work</td>
<td>55 °F to 95 °F</td>
<td>60 °F to 80 °F</td>
<td>NA</td>
<td>55 °F to 95 °F</td>
<td>Galvanized steel ASTM A653*</td>
<td>None</td>
</tr>
<tr>
<td>Return duct work</td>
<td>55 °F to 95 °F</td>
<td>60 °F to 80 °F</td>
<td>NA</td>
<td>55 °F to 95 °F</td>
<td>Galvanized steel ASTM A653*</td>
<td>None</td>
</tr>
</tbody>
</table>

This table is for information only and does not provide operational controls.


NA = not applicable.
A2.7.3 Health Protection System

The health protection system is designed to monitor and warn plant personnel of hazardous radioactive conditions that may occur as a result of malfunctions or accidents, provide contamination control, and provide limited computer-activated database management and status reporting. The health protection system employs radiation monitoring equipment and personnel contamination monitors.

The individual components of the health physics system are located throughout the CSB. Each instrument of the system has the ability to operate independently from the DCS. Not all of the health physics instruments are connected to the DCS. The data paths provided from the instruments to the DCS enable data collected by the instruments to be stored, compared, and displayed in one central location. The following is a list of the equipment designed for use in the CSB; and Figure A2-53, shows the locations of the monitoring equipment. The locations are based on radiation dose levels, operation occupancy factors, and egress pathways. These locations may change during facility operations if radiation protection determines that a different location will provide better protection for personnel.

- Multichannel Analyzer — One multichannel analyzer is located in the count room in the CSB. The multichannel analyzer is used for gamma spectrum analysis of radiological control swipes and air samples.

- Alpha/Beta Counter — Instruments in the countroom include a computer-controlled alpha/beta counting system that has a proportional detector and automatic sample changer. This system can simultaneously measure the alpha and beta activity on samples such as radiological control swipes and air samples, and print reports of the data.

- Germanium Detector — The germanium detector is used for gamma spectroscopy analysis of samples. The electronic signal from the germanium detector is analyzed by the multichannel analyzer indicating the gamma isotopes in the sample.

- Hand and Foot Monitor — One hand and foot monitor is located in the regulated change room (room 350). People place their feet in the foot slots and thrust their hands into the hand slots of the hand and foot monitor. The hands pressing in the hand slots activate the monitor, which measures radioactivity, if any, on the feet and hands. The chosen location ensures that personnel leaving the operating area can survey themselves before entering uncontaminated areas of the CSB.

- Area Radiation Monitors — Area radiation monitors, located throughout the CSB (Figure A2-53), monitor the general radiation level in the area surrounding their locations. A decade switch on the area radiation monitor provides for detection, indicating, and alarming for a range from 0.1 mrem/h to 10 rem/h (1 μSv/h to 100 mSv/h). All area radiation monitors have a microprocessor-based rate meter.
with readouts, including failure and high alarms with audible and visual annunciation. Area radiation monitors connect to the DCS through a remote terminal unit.

- Beta Continuous Air Monitors — Beta CAMs evaluate the amount of beta-emitting airborne radioactive material in sample air. Each beta monitor contains a Geiger-Muller detector and a count rate meter including failure and high radiation alarms. The beta CAMs are located in occupied areas having high potential for airborne contamination. For example, one is mounted on the MHM and another on the mobile containment service tent. Selected beta CAMs may be connected to the DCS. CAMs sample and monitor the air and are designed to provide early warning (local alarm bell and red rotating beacon light) of a significant release of radioactive material.

- Alpha Continuous Air Monitors — The alpha CAMs provide detecting, indicating, and alarming functions (local alarm bell and red rotating beacon light) for airborne alpha radioactive particulate. A pumped airflow is monitored and sampled for evidence of alpha airborne contamination. None of the alpha CAMs are connected to the DCS.

- Radioactive Gas Monitors — Radioactive gas monitors provide detection, indicating, and alarm functions for radioactive gases potentially released from MCOs. No radioactive gas monitors are connected to the DCS. These monitors are located on the MHM, tube vent and purge cart, and the mobile containment service tent.

- Body Frisker — A body frisker checks whole bodies for contamination. The body frisker monitors people for potential contamination as they exit the operating area airlock (room 25) and enter the uncontaminated areas of the support area.

- Record Air Samplers — Record air samplers, located throughout the CSB, provide a record of the work place air quality in high-occupancy and frequently traveled areas of the facility. A pumped airflow passing through high efficiency filters deposits particulate on the filter. These filters, counted using the alpha/beta counter and/or multichannel analyzer, can be used to sample room air for evaluation of potential inhalation hazards to personnel. The filters are exchanged by the Radiological Control personnel at a frequency specified by their management. These filters are exchanged if an incident involving airborne contamination occurs, because the filters provide an integrating sample over a period of time and CAMs provide immediate notification of a problem. These filters can be exchanged at the discretion of Radiological Control management to help evaluate an airborne contamination event. These data also can be used to support the concentrations recorded by a CAM alarm. The sample filter is counted by Radiological Control personnel and could be sent to the SNF Counting Facility for further analysis.
**Safety Considerations.** The health protection system is classified general service.

### A2.7.4 Airborne Emissions Monitoring System

The components of the CSB HVAC stack airborne emissions monitoring system include airflow rate and temperature measurements; a shrouded sampling probe for radioactive particulate and radionuclide monitoring; and two vacuum pumps (single redundancy), which draw the effluent sample at a flow rate that is nominally set at 2 standard ft³/min through a splitter. The sample splitter block and flow measurement and control components distribute, measure, and control the flow of the sample from the sample probe to the samplers. The splitter block is designed to provide a uniform representative sample to each sample collector assembly. There is a mass flow controller in each of the two flow streams that maintains 1 standard ft³/min of flow through each of the two streams. One stream contains a record air sampler and the other stream contains the alpha/beta detector. The system is designed to the following criteria:


- Monitor radionuclides that contribute 0.1 mrem/yr ($1.0 \times 10^{-3}$ mSv/yr) or more to the anticipated unabated offsite effective dose ($^{238}$Pu, $^{239}$Pu, $^{240}$Pu, $^{241}$Pu, $^{241}$Am, $^{137}$Cs, and $^{90}$Sr) (this criterion is based on WAC 246-247)

- Monitor radionuclides that contribute 25% or more of the anticipated abated offsite effective dose ($^{241}$Am) (this criterion is based on WAC 246-247)

- Provide constant air monitoring of alpha/beta particulate (this criterion is based on DOE Order 5400.1)

- Operate such that CAM downtime for scheduled maintenance does not exceed 8 hours in each 24-hour period; unscheduled downtime not greater than 72 hours in each 7-day period
• Provide for a minimum collection efficiency of 50% or more for 10 μm sized particles.

Figure A2-53 shows the layout of the monitoring equipment. Figure A2-52 shows the stack monitoring equipment. Each sample flow channel of the system is controlled by an independent microprocessor with a self-contained rate meter that will continue to operate even if the DCS or its communication lines fail. Local indications such as count rate data and radiation field intensities are available from each channel. If alarm conditions exist, local visual and audible annunciation occurs at the monitor panel independent of the DCS. Each monitoring channel is capable of self-testing using an independent radioactive check source. Testing can be initiated locally. The complete system provides monitoring, database management, and status reporting.

The remote stack sampler uses a shrouded probe located at an elevation of 25 ft, 9.25 in. The probe is installed with the nozzle at the cross-sectional center of the stack. The flow from this probe is piped to the in-line sample splitter and then to the detectors and record sampler located in the sampler skid. All piping is of seamless construction to reduce plateout of radioactive particulate and prevent hot spots. The alpha/beta and gamma record sampler collects a sample of particulate and gas exiting the stack for subsequent radionuclide analysis. The line loss and filter efficiency are such that the collected particle sample contains >50% of the 10 μm and larger (aerodynamic equivalent diameter) particles that are present in the free system. The alpha/beta scintillation detector, as described in NRC MD-345, *Iodine Detector*, monitors alpha activity from 239Pu and 241Am in stack effluent particulate. The detector also monitors for 90Sr beta emissions in stack particulate. The alpha/beta detector monitors for real-time early warning of an out-of-ordinary emission to facility personnel. The detector has circuitry for background subtraction of radon-thoron activity. Test ports have been installed on the exhaust stack (two near the base, two near the lower sample port, and two near the upper flow port) for manual monitoring of stack flow using portable flow measuring equipment and/or manual samplers.

The electronics console located in the CAEM room controls all functions of the HVAC stack monitoring system. A digital rate meter, bar code reader, pump control module, data storage, and transfer computer are included in this unit. All monitoring information that is provided and/or available to the CSB DCS comes from this unit, with the exception of the stack flow signal. The alarm setpoints can be set and changed from the console.

Item 25 of the NRC equivalency requirements (HNF-SD-SNF-DB-003) involves reviewing federal effluent monitoring requirements, for example, Title 10, *Code of Federal Regulations*, Part 20, “Standards for Protection against Radiation” (10 CFR 20); Title 10, *Code of Federal Regulations*, Part 70, “Domestic Licensing of Special Nuclear Material,” Section 70.59, “Effluent Monitoring Reporting Requirements” (10 CFR 70); and 10 CFR 835, to ensure the necessary monitoring instrumentation has been provided. The following is a summary of the review performed.
1. 10 CFR 20 provides standards for protection against radiation; Section 20.1501(b) discusses the need to calibrate equipment used for effluent monitoring. More prescriptive requirements on emission monitoring are found in 40 CFR 61 and in WAC 246-247. The radioactive airborne emission control system and monitoring system of CSB are in compliance with these requirements and have been approved by the Environmental Protection Agency, Region 10, and the Washington Department of Health (note the approved permit application, DOE/RL-98-30, describing the emissions control and monitoring system).

2. 10 CFR 70.59 provides for effluent monitoring reporting requirements. A review of 10 CFR 70.59 shows it requires the facility to provide NRC semiannual effluent reports. There are no liquid effluent streams for the CSB, so the only effluent stream is the airborne emissions stream. Annual emissions reporting to the Environmental Protection Agency is required by 40 CFR 61, Subpart H, and annual emissions reporting to the Washington Department of Health is required by WAC 246-247.

3. 10 CFR 835 does not apply because it has requirements for occupational radiation protection and not for environmental protection or monitoring of facility airborne emissions.

Safety Considerations. The airborne emissions monitoring system is classified general service.

A2.7.5 Inert Gas Supply System

The inert gas supply system (helium) interfaces with the cask servicing system (abnormal operations), the MCO tube vent and purge cart system, and sampling/weld stations. The inert gas supply system contains gas supply banks with an initial pressure of at least 2,000 lb/in² gauge. The gas bank is certified by the vendor upon shipment to ensure administrative control of gas content and purity. The gas supply pressure is reduced to a nominal pressure of 120 lb/in² gauge for distribution to the cask receiving pit and sampling/weld stations. Quick-disconnect fittings near the cask receiving pit and the sampling/weld stations provide a helium supply for purging casks and sampling MCOs. Flexible hoses from the MCO servicing system and the sampling/weld equipment are attached to these fittings when helium is needed.

The inert gas supply system is designed to provide four main functions for MCO servicing (abnormal operations): (1) high-pressure purging of an MCO cask in the cask receiving pit; (2) controlling dilution for gases vented from a cask in the cask receiving pit; (3) refilling an MCO at the sampling/weld station; and (4) controlling dilution for the gas vented from the MCO at the sampling/weld station to the HVAC exhaust duct to reduce the possibility of hydrogen combustion in accordance with NFPA 69, Explosion Prevention Systems. A pressure control valve automatically regulates the amount of inert gas used to dilute the concentration of hydrogen present in the MCO vent gas based on the pressure present in the MCO vent gas line.
The inert gas supply system is designed to provide inert gas to the tube vent and purge cart at the inert gas fill station. This fill station provides a minimum of 2,000 lb/in² gauge inert gas to refill the two inert gas cylinders present on the tube vent and purge cart. Helium is also supplied at reduced pressure to the sampling/weld station for refilling of monitored MCOs and for leak testing after welding cover cap assemblies to MCOs. The tube vent and purge cart bottles provide only enough capacity for one MCO storage tube volume of gas. The tube vent and purge inert gas fill station provides a quick and efficient method of refilling the tube vent and purge cart cylinders for reuse. The inert gas system piping and components located inside the CSB support and operations area are designed to ANSI/ASME B31.3, Process Piping, for the safety-significant portions that connect to the MCO servicing system and to ASME B31.1, Power Piping, for the general service portions that supply inert gas to the vent and purge cart refill station and to the helium let-down station. All pressurized gas cylinders received at the CSB have a cap screwed on the cylinder that prevents damage to the cylinder valve during shipment and handling. This cap remains on the cylinder until after the cylinder is chained to the cylinder rack and personnel are ready to attach the high-pressure discharge piping to the cylinder valve. All pressurized gas cylinders received at the CSB have a cylinder valve. According to ANSI/CSA/CGA, Standard V-1, Compressed Gas Cylinder Valve Outlet and Inlet Connections, a maximum hole size in the inlet to the butt of the cylinder valve restricts airflow through the valve and minimizes propulsion effects should the valve be accidentally severed at the top of the cylinder. Both the cylinder cap and the cylinder valve protect a pressurized gas cylinder from causing serious propulsion effects.

Figures A2-24, A2-42, and A2-45 show the inert gas supply system general service lines supplying inert gas to the load-in/load-out cask receiving pit, the tube vent and purge cart refill station, and to the helium let-down station in the sampling/weld area. A low pressure (125 lb/in² gauge) supply line is routed through the south side of the compressor area. There are two doorways separating this line from the control room (approximately 60 ft) where personnel are stationed. This is sufficient distance to protect personnel should a helium leak occur in the supply line. Also, the support area ventilation system will dilute and evacuate an accidental release of helium from this area. Pressure relief devices (valve and rupture disks) are located in the sampling/weld area and are mounted on the south wall. PSV-728 and PSE-1 provide pressure relief if the helium supply pressure exceeds 135 lb/in² gauge or 150 lb/in² gauge, respectively. Table A2-12 lists the operating conditions for which the inert gas supply system has been designed.

Safety Considerations. The helium supply system rupture disk (PSE-1) is classified safety significant and important-to-safety Category B. The remainder of the helium supply system is classified general service.
### Table A2-12. Inert Gas Supply System Design and Operating Conditions.

<table>
<thead>
<tr>
<th>Component</th>
<th>System design conditions</th>
<th>System operating conditions</th>
<th>Design operating conditions</th>
<th>Operating environment</th>
<th>Material</th>
<th>Corrosion allowance</th>
</tr>
</thead>
<tbody>
<tr>
<td>Piping high pressure</td>
<td>-20 °F to 150 °F,</td>
<td>60 °F to 85 °F,</td>
<td>60 °F to 85 °F</td>
<td>55 °F to 95 °F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td></td>
<td>2,500 lb/in² gauge</td>
<td>2,000 lb/in² gauge</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Valves</td>
<td>-20 °F to 150 °F</td>
<td>60 °F to 85 °F</td>
<td>60 °F to 85 °F</td>
<td>55 °F to 95 °F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td>Pressure safety valves and</td>
<td>High</td>
<td>60 °F to 85 °F</td>
<td>60 °F to 85 °F</td>
<td>55 °F to 95 °F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td>ruptured disks</td>
<td></td>
<td>2,500 lb/in² gauge</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>125 lb/in² gauge</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Low</td>
<td>125 lb/in² gauge</td>
<td>120 lb/in² gauge</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Piping low pressure</td>
<td>-20 °F to 150 °F,</td>
<td>60 °F to 85 °F,</td>
<td>60 °F to 85 °F</td>
<td>55 °F to 95 °F</td>
<td>Carbon steel</td>
<td>NA</td>
</tr>
<tr>
<td></td>
<td>175 lb/in² gauge</td>
<td>120 lb/in² gauge</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

This table is for information only and does not provide operational controls.

NA = not applicable.
A2.7.6 Seismic Detection System

Redundant accelerometers are mounted on the northeast curb of the load-in/load-out area and are designed to detect building motion and dynamic response caused by a seismic event. The accelerometer control panel contains electrical connections to power sources for both the MHM and receiving crane. When excess seismic motion is detected by the accelerometers (the channel trip setting is 0.19 g peak ground acceleration, which is approximately 54% of the maximum horizontal or vertical spectra for the 0.35 g earthquake with 7% damping), a power disconnect switch for the receiving crane actuates causing loss of power. On loss of power, the receiving crane load is locked in the current position preventing further movement of the load or crane.

The accelerometers, upon detecting a seismic event, also activate a disconnect switch for the MHM power. The accelerometers are normally energized to hold closed the power supply contacts for the MHM. A seismic event will de-energize the accelerometers which in turn de-energizes the MHM. A loss of power for the MHM actuates seismic clamps for both sides of the bridge, stops turret rotation, and locks the shield plug hoist and the MCO hoist in the current positions. These clamps and locking features are designed to prevent further motion of the MHM, minimize damage to the MCO, and prevent a potential contamination spread.

The accelerometer panel and the accelerometer power panel are installed on an equipment rack that has been seismically qualified according to CSB-S-0064, *MHM and Receiving Crane Seismic Trip Supports*. Mounting on a seismically qualified equipment rack provides assurance that an earthquake can be detected with subsequent safe shutdown of the receiving crane and MHM.

Safety Considerations. The seismic detection system for the MHM, the receiving crane, and the power-disconnect system for the MHM are classified general service.

A2.8 UTILITY DISTRIBUTION SYSTEMS

This section provides a schematic outline of the basic utility distribution systems, including a description of the offsite power supplies and onsite components of the system. The CSB design relies on offsite power and road transport of bulk gases and supplies for normal operation. The CSB basic utility distribution systems, including the UPS, ensure continued operation of safety systems during NPHs or offsite emergencies.

A2.8.1 Normal Electrical Power

The normal electrical distribution system is shown in Figure A2-54. The figure shows the one-line connections between Hanford Site power, the CSB transformers, and switchgear for...
voltage and load control during normal operation of the CSB. The normal electrical distribution system comprises the following subsystems.

- The 13.8 kV, 3-phase Hanford Site overhead utility system brings power to a utility pole located northeast of the CSB through a cutout switch and fuses, and terminates at the primary circuit of the SNF CSB pad-mounted transformer.

- The pad-mounted transformer converts 13.8 kV to 480Y/277 V, which is then supplied through the utility metering cabinet to the low-voltage switchgear cabinet SG-32-202 (room 031) where it is distributed through circuit breakers to four MCCs.

- MCC-33-207, MCC-33-208, MCC-32-209, and MCC-32-210 distribute 480Y/277/208/120 V (ac) power to distribution panels A, B, D, G, H, and J. Each circuit is controlled by a circuit breaker designed to protect the circuit components if an electrical fault should occur. Distribution panels A, B, D, G, and H are located in the electrical equipment room (room 031): panels A, B, and D are on the west wall, and panels G and H are on the south wall. Distribution panel J receives power through MCC-32-210 and distributes power to the sampling/weld station equipment and distribution panel K, a 208/120 V (ac) panel. Panels J and K are located on the east wall of the sampling/weld station area.

- The UPS receives 480 V (ac), 3-phase, 60 Hz power from the dedicated MCC MCC-33-207. This MCC is part of the normal electrical distribution system and is considered the normal power supply for the UPS. If the normal power supply is unable to supply power, then the backup battery will automatically supply uninterrupted power until power from a portable diesel generator or normal power supplies are available. Distribution panels C and F receive 208/120 V (ac), 3-phase, 60 Hz power from the UPS. Power to distribution panels C and F may also be connected to either MCC-33-207 or MCC-32-210 by closing the appropriate circuit breaker on UPS-33-213. The UPS and panels C and F are located in the UPS room (room 020). Portable diesel-powered electrical generators are available onsite and can be brought to the CSB to supply electrical power.

Equipment support racks containing control panels, transfer switches, conduit supports, and circuit breakers have been designed using anchor bolts and unistrut construction to match the safety-class criteria of the equipment.

Cutout switch DD-32-101 is supplied electrical power from an existing 13.8 kV overhead utility circuit. The fused cutout switch provides electrical isolation during planned maintenance activities or circuit overloads. Each fuse installed in this switch assembly is rated for 50 amps. The cutout switch has ganged switching components that ensure all three phases are interrupted whenever the switch is placed in the open position. Power from the cutout switch is routed underground to the primary circuit of the pad-mounted transformer. Cable that has been pulled
through conduit meets the requirements in NFPA 70, *National Electrical Code*, for this type of
cable. Electrical conduit supports for safety significant and general service installations have been
designed and installed for the electrical cable trays.

Pad-mounted transformer XT-32-102 is energized by 13.8 kV through its 200 amp primary
disconnect switch and fuses. This transformer is rated for 1,000 kVA and steps 13.8 kV down to
480/277 V. It is a self-cooled unit with heat ratings of 65 °C (149 °F) temperature rise above
40 °C (104 °F) ambient. The transformer secondary circuit provides 480/277 V power to the
utility metering cabinet. Four grounding rings (loops) around this transformer ensure safe
grounding for the life of the facility. These grounding rings are electrically bonded to the
transformer and the CSB ground system.

Utility metering cabinet UC-32-201 houses circuit monitoring equipment and a three-phase
main circuit breaker. A solid-state, kilowatt-hour-meter/watt-hour-meter-with-demand meter
mounted on the door of the metering cubicle receives input from potential transformers and
current transformers housed in the metering compartment measuring power usage of the SNF
CSB. The utility metering cabinet circuit breaker isolates the pad-mounted transformer power
from the indoor switchgear. The circuit breaker is a 600 V, 1,200 amp frame with a 100,000 amp
interrupting capability. The 1,200 amp solid state trip unit has an adjustable long-time pickup,
long-time delay, short-time pickup, and short-time delay. Instantaneous override in conjunction
with the built-in ground fault protection is provided. When the main utility metering cabinet
circuit breaker is closed, it provides 480/277 V power to the low-voltage indoor switchgear.

Low-voltage switchgear cabinet SG-32-202 is constructed to house eight 3-phase circuit
breakers. Each breaker compartment is 92 in. high and 60 in. deep, and is isolated from other
breaker compartments with grounded metal barriers. The compartment doors are designed to
remain closed when racking the breakers to any of their three positions: disconnected, test, and
connected. A positive mechanical interlock is provided to prevent the breakers from being racked
in or out unless the breaker is tripped and to prevent the breaker from being closed while it is
being racked in or out. Each breaker compartment is provided with a pilot light for breaker open
or breaker closed. Four circuit breakers provide normal electrical power and circuit protection to
the four MCCs in the CSB. The remaining two circuit breakers and two empty cubicles are
spares for future use.

MCC-33-207, MCC-33-208, MCC-32-209, and MCC-32-210 are energized by the normal
electrical distribution system from low-voltage switchgear cabinet SG-32-202. All four MCCs are
part of the 480/277 V (ac), 3-phase, 4-wire, 60 Hz, grounded distribution system. MCC cabinets
are standard, prefabricated, vertical sections comprising individual control units arranged to
provide a completely dead-front, totally enclosed assembly. All motor starters are of the full-
voltage, nonreversing type. Each combination starter is housed in an individual compartment
removable from the front by disconnecting the necessary control and load leads. Each
compartment door is provided with an interlocking feature to prevent the opening of any door
when the unit it houses is energized. Each MCC is equipped with an ammeter to monitor total
MCC loads. Power distribution from the MCCs is divided into two basic groups. One group
supplies individual loads with optional control circuits, and the other group supplies the
distribution panels (A through H) where smaller loads are subdivided into single load groups.

Distribution panels A, C, D, F, and J receive 480/277 V (ac) 3-phase power from their
MCC and distribute it through individual molded case circuit breakers that are mounted in each
panel. Distribution panels B, E, G, H, and K are supplied power from their MCC through a
dedicated indoor dry transformer, which steps the voltage down to 208/120 V, 3-phase, 4-wire
for distribution through individual molded case circuit breakers. All panel assemblies are enclosed
in a surface-mounted steel cabinet. UPS distribution panels C and F have the ground bus
electrically isolated from the enclosure; all other assemblies have a ground bus bolted to the frame
and an isolated neutral bus. In compliance with 29 CFR 1910.306, the power supply
(DA-33-217, Figure A2-54, sheet 6) for the sampling/weld gantry crane can be viewed by
personnel, no fixed working platform can be attached to the crane for maintenance and the crane’s
power supply breaker is labeled on distribution panel J.

Lighting systems for the CSB comply with Illuminating Engineering Society of North
America’s Lighting Handbook (IES 1987) and ASHRAE 90-75R, Energy Conservation in New
Building Design. The lighting systems include exterior, interior, standby, and emergency lighting.
Exterior lighting illuminates the outside of the buildings using wall-mounted low-pressure sodium
light fixtures, which are controlled by light sensitive photocells in compliance with DOE
Order 5400.1, Section 1650-1. Interior lighting consists of normal lighting and standby lighting.
Illumination levels follow DOE Order 6430.1A, which recommends 50 footcandles for
workstation lighting, 30 for work area lighting, and 10 for non-work areas. Fluorescent light
fixtures with energy saving ballasts are used in the support building rooms. High intensity
discharge lighting fixtures are used in the operating area shelter. Standby lighting systems contain
rechargeable batteries providing power during power failures. The interior and standby lights are
controlled using manual light switches. Emergency lighting consists of individual lighting fixtures
powered by an integral battery and self-luminous exit signs which comply with NFPA 101 and the
life safety code, DOE Order 6430.1A, Section 1635-1. The emergency lighting fixtures are
installed in areas essential to life safety, ingress and egress routes from the equipment areas.
Portable lighting may be needed during inspection of the cask surface to meet the requirements of
ANSI/ANS-57.2-1992 (100 footcandles).

Table A2-13 shows the operating conditions for which the electrical systems have been
designed.

Safety Considerations. The normal electrical power system is classified as general service.
Table A2-13. Electrical Room Design and Operating Conditions.

<table>
<thead>
<tr>
<th>Component</th>
<th>System design conditions</th>
<th>System operating conditions</th>
<th>Design operating environment</th>
<th>Operating environment</th>
<th>Material</th>
<th>Corrosion allowance</th>
</tr>
</thead>
<tbody>
<tr>
<td>Electrical equipment</td>
<td>NA</td>
<td>NA</td>
<td>55 °F to 104 °F</td>
<td>55 °F to 95 °F</td>
<td>Manufacturer standard</td>
<td>NA</td>
</tr>
</tbody>
</table>

This table is for information only and does not provide operational controls.

NA = not applicable.
A2.8.2 Backup Electrical Power System

The backup power system designed for the CSB will not be installed and connected at this time. No safety-class or safety-significant SSCs that would require a backup power system have been identified in Chapter A3.0. According to HNF-4776, Canister Storage Building Compliance Assessment, SNF Project NRC Equivalency Criteria — HNF-SD-SNF-DB-003, this meets the NRC equivalency requirement to evaluate loss of AC power to the facility (4). Two rooms in the support area building (rooms 18 and 34) are designated as utility rooms; and the two diesel supply tanks on the north sides of these rooms will not be used.

A2.8.3 Uninterruptible Power System

UPS UP-33-213 (Figure A2-54, sheet 7) is rated at 20 kVA, 208/120 V (ac), 3-phase, 4-wire, 60 Hz output and 480 V (ac), 3-phase, 60 Hz input. UP-33-213 functions to supply uninterrupted, reliable power to SNF CSB instrumentation distribution panels DA-33-304 (Panel F) and DA-33-303 (Panel C). MCC-32-207 is the normal power supply. If the normal power supply is unable to supply power, backup battery BA-33-301 automatically starts supplying power until a portable diesel generator or normal power is available, or until the backup battery power is depleted (about 30 minutes at full load). If the UPS should fail in such a manner that neither the normal power supply nor the UPS battery could supply power, the UPS maintenance bypass switch would be positioned to bypass the failed components and supply alternate power from MC-32-210 to the UPS.

The UPS is of a solid-state design, housed in a steel cabinet. The three-phase, four-wire system incorporates an ungrounded neutral bus and a ground bus that is electrically isolated from the enclosure. The major components comprise a rectifier-charger, battery and cabinet, inverter, static transfer switch, maintenance bypass switch, bypass transformer, synchronizing equipment, system control panel, metering, alarms, and protective devices. A UPS alarm is connected to the DCS and alerts personnel when a failure occurs in the UPS.

The rectifier-charger is a solid-state unit providing direct current to the inverter and to the battery for charging. The rectifier-charger unit has input current limiting set at the factory to 115%. An output filter is incorporated with the rectifier-charger unit to minimize ripple current into the battery. Under normal conditions, ripple current is limited to less than 2% root mean square. The charging rate of the rectifier-charger is sufficient to restore the battery from discharge to 95% charge within 8 hours.

The output frequency of the inverter is controlled by an oscillator that is temperature compensated, and is adjusted from 5% of the rated frequency to 0.1% for both steady state and transient conditions. The inverter output is designed to stay synchronized with the static bypass line provided the static bypass line remains within ±3 Hz of the nominal frequency. If the inverter frequency goes outside these limits, the inverter is designed to break synchronization with the lines and run on its internal frequency. The inverter is designed to sustain an overload across its...
output terminals up to 150% load, while supplying any load within its rating, without reducing the
output voltage. The efficiency of this unit is 85% at rated load. When the load is on the
maintenance bypass line, it is possible to check the operation of the rectifier-charger, inverter,
static transfer switch, and battery.

The battery cells are lead-calcium and contained in plastic vessels that have one vent for
each cell. The capacity of the battery is designed to supply direct current to the inverter for an
outage period of 30 minutes with the inverter operating at full rated output. A battery circuit
transfer breaker is provided.

**Safety Considerations.** The UPS is classified general service.

### A2.8.4 Distributed Control System

The DCS is designed to serve as the central monitoring and control system for CSB facility
conditions, systems (particularly the health physics monitoring system), and processes (specifically
the inert gas dilution of vented cask gases). Two central processing units provide redundant data
collection from instrumentation and monitoring systems located throughout the facility.

Components in the field (e.g., instruments, actuators, sensors) are connected to the DCS through
two remote termination unit cabinets that are located in rooms 019 and 030 in the facility. A data
highway links all components of the DCS together. A color printer in the control room permits
printing of hardcopy listings of alarms, control actions, and other reports by request. An audible
alarm in the control room alerts personnel when setpoints are exceeded. The DCS is shown on
Figure A2-55 and the instruments are listed in Table A2-14.

Two work stations provide monitoring of components connected to the DCS, control
functions, and the means for archiving mass data for safe storage. A programmable logic
controller accepts sensor inputs from field devices, accepts control inputs from the work stations,
processes these inputs through "ladder logic" algorithms in the programmable logic controller
processor, then generates and transmits control signals to field devices such as controllers or
alarms, and returns to the DCS for data recording or alarm annunciation. Programmable logic
controller remote input/output units concentrate and transmit data to another data processing
unit, which then acts as controller for the data processing unit represented by the input/output
unit. The DCS receives electrical power from the normal electrical distribution system through
battery-backed UPS DA-33-213.

The DCS provides monitoring capabilities for systems that provide for confinement of
incidental radioactive contamination to the CSB operating area by supporting key equipment
connected to health physics monitoring systems, the HVAC system, the MCO servicing system,
and the vault cooling system. For the vault cooling system, the air inlet and exhaust temperatures
and the flow rate are monitored using the DCS. The only control associated with the DCS is the
MCO vent inert gas dilution control valve in the cask servicing system. All other functions of the
DCS are monitoring functions of safety-significant equipment.
Table A2-14. Distributed Control System Instrument Interface.

<table>
<thead>
<tr>
<th>System</th>
<th>Analog input</th>
<th>Analog output</th>
<th>Discrete input</th>
<th>Discrete output</th>
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<tbody>
<tr>
<td>Health physics stack monitoring</td>
<td>FI-281</td>
<td>RI-405</td>
<td>RA-410</td>
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<td>FI-408</td>
<td>RI-406</td>
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<tr>
<td></td>
<td>RI-404</td>
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<td></td>
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</tr>
<tr>
<td>Health physics support area</td>
<td>ARM-510</td>
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<td>CAM-560</td>
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<tr>
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<td>ARM-521</td>
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<td>RAS-602</td>
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<td>RAS-605</td>
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<td>Health physics operating area</td>
<td>ARM-513</td>
<td>ARM-525</td>
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<tr>
<td></td>
<td>ARM-514</td>
<td>ARM-526</td>
<td>ARM-528</td>
<td></td>
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<tr>
<td></td>
<td>ARM-515</td>
<td>ARM-527</td>
<td>ARM-529</td>
<td></td>
</tr>
<tr>
<td></td>
<td>ARM-518</td>
<td>ARM-528*</td>
<td>ARM-530</td>
<td></td>
</tr>
<tr>
<td></td>
<td>ARM-519</td>
<td>ARM-529*</td>
<td>ARM-531</td>
<td></td>
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<tr>
<td></td>
<td>ARM-521</td>
<td>ARM-530</td>
<td>ARM-532</td>
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<td>CAM-563</td>
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<td></td>
<td>ARM-523</td>
<td>ARM-533</td>
<td>(alarm)</td>
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<td>CAM-568</td>
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<td>RAS-608</td>
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<tr>
<td>Air compressor</td>
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<td>CA-241</td>
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<tr>
<td>Air dryer</td>
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<td>CA-251</td>
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<td>HVAC pump</td>
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<td>LAH-119</td>
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<tr>
<td>UPS</td>
<td></td>
<td></td>
<td>UPS alarm</td>
<td></td>
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<td>Vault intake flow</td>
<td>FT-285-1</td>
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<td>Vault inlet temperature</td>
<td>TI-286-1</td>
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<td>Vault outlet temperature</td>
<td>TI-299-1</td>
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<tr>
<td>MCO servicing</td>
<td>PIC-115</td>
<td>TIT-109</td>
<td>PY-115</td>
<td></td>
</tr>
<tr>
<td>MCO sampling station</td>
<td>PI-721</td>
<td>TI-723</td>
<td></td>
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</tr>
<tr>
<td></td>
<td>PI-723</td>
<td>TI-731</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

*a* High radiation alarm (stack).

*b* Equipment fail alarm (stack monitoring).

HVAC = heating, ventilation, and air conditioning.

MCO = multi-canister overpack.

UPS = uninterruptible power supply.

- = not applicable.

* = portable.
Safety Considerations. The DCS is classified general service.

A2.8.5 Instrument and Plant Air System

The CSB instrument and plant air system comprises two 2-stage, oil-free, air-cooled, rotary screw compressors operating in alternating lead-starting fashion to deliver a minimum of 200 standard ft³/min of compressed air at 125 lb/in² gauge at the compressor discharge. The compressors have self-contained noise abatement features in compliance with DOE Order 6430.1A, Section 1300-12.4.3. Compressor operation is controlled by a separate microprocessor which senses pressure from instrument air receiver PT-248. When PT-248 senses a compressed air pressure drop to 90 lb/in² gauge, it signals the microprocessor to start a compressor. If pressure does not start rising, the second compressor also starts. The microprocessor is programmed to maintain the air receiver at a 100 ± 10 lb/in² gauge setpoint. The microprocessor will alternate the lead compressor based on a programmable time duration, presently set at 20 hours. The backup compressor will normally remain shut down in an "auto-start" mode, and start only if the lead compressor cannot supply the required air demand. Should that happen, both compressors will run. With no air demand, and after preset events, both compressors will stop and remain stopped in "auto-restart" mode. Subsequent air demand (PT-248 senses < 90 lb/in² gauge) will cause the current lead compressor to run. PIT-248 will signal the compressor selected as lag to start.

Compressed air from the compressors passes through a coalescing prefilter, a heatless desiccant air dryer, and a particulate after filter before it collects in a 340 ft³ air receiver to provide dry, oil- and particulate-free air on demand to the compressed air system, which supplies the instrument air subsystem and the service air subsystem. The instrument air subsystem distributes compressed air at 0 to 20 lb/in² gauge to HVAC equipment (pneumatic actuators, control valves, dampers, and instruments) throughout the CSB and the MCO service station area. The service air subsystem distributes compressed air at 100 to 110 lb/in² gauge to the MHM festoon supply, the weld stations, and for general use (e.g., for air tools) to utility stations in the north end of the CSB near the receiving area, as shown on Figure A2-56.

The two drying towers of air dryer DR-1 remove moisture from the air discharged from the instrument air compressors by passing it over an activated alumina desiccant material with one tower operating in drying mode while the other tower is in regeneration mode. When the desiccant is saturated, the active drying tower is automatically switched from a drying mode to regeneration mode. The other drying tower is switched from regeneration mode to drying mode. The dryer is a "heatless regenerative" dryer, meaning that regeneration of the desiccant is accomplished by routing a small amount of hot, dry, compressed air from the discharge of the active drying tower in a counter-current flow through the regenerating column to drive the moisture off the surface of the desiccant material rather than by using electric heating elements to provide the heat.
Instrument air receiver VX-1 is sized to provide up to 40 minutes of instrument air to normal instrument air loads in the event of a power loss to provide for shutdown of the systems using instrument air (e.g., HVAC, MHM). Receiver VX-1 is protected from overpressurization with pressure relief valves whose discharges are routed to safe areas away from any possible impingement on personnel. This compressed air is for plant uses only and is never to be used as breathing air. Table A2-15 lists the operating conditions for which the instrument air system has been designed.

Safety Considerations. The instrument and plant air system is classified general service.

A2.8.6 Liquid Waste Collection

The liquid waste collection system is designed to collect water condensate from HVAC air handling unit cooling coils and instrument air compressors, and provide for transfer of the condensate to approved containers for disposal. Water draining from these systems collects in sumps, and an air-operated pump transfers water from the sumps into the disposal container (Figure A2-57). Sumps collect the water because there are no floor drains in the facility. This system has been designated general service. The system is designed for a condensate collection rate of approximately 1 gal/h for 8 hours per day. The liquid waste collection system is not a floor drain or fire suppression water catchment. A concrete sealer has been applied to the walls and bottom of each sump that protects and waterproofs the concrete.

Sump SU-1 is a 4 ft by 2 ft by 2 ft deep recessed area in the support area building floor between the HVAC air handling units, as shown on Figure A2-53. A second sump (SU-2), 2 ft by 2 ft by 2 ft deep, is located between the instrument and plant air compressors, as shown on Figure A2-53. Both sumps are fall-protected with metal floor gratings.

HVAC condensate drains to the liquid collection sump SU-1 from drip pans located in the HVAC equipment room. Water separated by the air dryer system filters and compressed air system air-water separators drains into sump SU-2. An alarm is activated on the DCS when the level sensor in sump SU-1 indicates that the water level has reached the alarm setpoint (half full for the sump). When the alarm is received, personnel position a container and a portable pump at the sump, positions the pump suction and discharge hoses, connects the pump to a service air supply, and pumps the sump contents into the drum. According to DOE/RL-97-67, Pollution Prevention and Best Management Practices Plan for State Waste Discharge Permits ST4508 and ST4509, sampling and analysis of the drum contents will only be performed on the first few drums to document that the discharge meets Washington State Waste Discharge Permit ST 4509 for discharge to the ground. If it is suspected during normal operations that the condensate has been contaminated radiologically or chemically, it will be sampled and analyzed. When the contents are released for disposal, the container may be transported to a facility at which the contents can be transferred to the Treated Effluent Disposal Facility. If the contents meet the criteria for discharge under Washington State Waste Discharge Permit ST 4509 (DOE/RL-97-67) for cooling.
Table A2-15. Compressed and Instrument Air Design and Operating Conditions.

<table>
<thead>
<tr>
<th>Component</th>
<th>System design conditions</th>
<th>System operating conditions</th>
<th>Design operating environment</th>
<th>Operating environment</th>
<th>Material</th>
<th>Corrosion allowance</th>
</tr>
</thead>
<tbody>
<tr>
<td>Compressors (CX-1A and CX-1B)</td>
<td>125 lb/in² gauge, 55 °F to 120 °F</td>
<td>105 to 125 lb/in² gauge, 60 °F to 120 °F</td>
<td>5% to 95% RH, 60 °F to 85 °F (normal), 5 °F to 115 °F (extremes)</td>
<td>5% to 95% RH, 55 °F to 95 °F</td>
<td>Carbon steel, cast iron</td>
<td>None</td>
</tr>
<tr>
<td>Prefilter (F-5)</td>
<td>150 lb/in² gauge, 120 °F (maximum), -40 °F dew point</td>
<td>105 to 125 lb/in² gauge, 60 °F to 120 °F</td>
<td>5% to 95% RH, 60 °F to 85 °F (normal), 5 °F to 115 °F (extremes)</td>
<td>5% to 95% RH, 55 °F to 95 °F</td>
<td>Carbon steel, cast iron</td>
<td>None</td>
</tr>
<tr>
<td>Air dryer (DR-1)</td>
<td>150 lb/in² gauge, 120 °F (maximum)</td>
<td>105 to 125 lb/in² gauge, 60 °F to 120 °F</td>
<td>5% to 95% RH, 60 °F to 85 °F (normal), 5 °F to 115 °F (extremes)</td>
<td>5% to 95% RH, 55 °F to 95 °F</td>
<td>Carbon steel, ASTM A515, ASME VIII, Division 1</td>
<td>0.0625 in.</td>
</tr>
<tr>
<td>Air receiver (VR-1)</td>
<td>150 lb/in² gauge, 250 °F</td>
<td>125 lb/in² gauge, 60 °F to 120 °F</td>
<td>5% to 95% RH, 60 °F to 85 °F (normal), 5 °F to 115 °F (extremes)</td>
<td>5% to 95% RH, 55 °F to 95 °F</td>
<td>Carbon steel, ASTM A515, ASME VIII, Division 1</td>
<td>0.125 in.</td>
</tr>
<tr>
<td>Plant compressed air piping</td>
<td>175 lb/in² gauge, -20 °F to 150 °F</td>
<td>105 to 125 lb/in² gauge, 60 °F to 120 °F</td>
<td>5% to 95% RH, 60 °F to 85 °F (normal), 5 °F to 115 °F (extremes)</td>
<td>5% to 95% RH, 55 °F to 95 °F</td>
<td>Carbon steel, bronze</td>
<td>0.065 in.</td>
</tr>
<tr>
<td>Instrument air piping</td>
<td>175 lb/in² gauge, -20 °F to 150 °F</td>
<td>20 or 85 lb/in² gauge, 60 °F to 120 °F</td>
<td>5% to 95% RH, 60 °F to 85 °F (normal), 5 °F to 115 °F (extremes)</td>
<td>5% to 95% RH, 55 °F to 95 °F</td>
<td>Carbon steel, bronze</td>
<td>0.065 in.</td>
</tr>
<tr>
<td>Service air piping</td>
<td>175 lb/in² gauge, -20 °F to 150 °F</td>
<td>100 or 110 lb/in² gauge, 60 °F to 120 °F</td>
<td>5% to 95% RH, 60 °F to 85 °F (normal), 5 °F to 115 °F (extremes)</td>
<td>5% to 95% RH, 55 °F to 95 °F</td>
<td>Carbon steel, bronze</td>
<td>0.065 in.</td>
</tr>
</tbody>
</table>

This table is for information only and does not provide operational controls.


RH = relative humidity.
water and condensate, the condensate may be used, as a best-management-practice, for lawn
irrigation or dust abatement. CSB Operations management is responsible for determining
disposition of container contents that are found to be in excess of Treated Effluent Disposal
Facility and/or State Waste Permit release limits.

Sumps in the FFTF pit and the MHM maintenance pit (one in each pit, 18 in. x 18 in. x
18 in. deep) collect fluids that drain from the equipment placed in these pits. The sumps may
contain organic fluids (crane lubricating oils or hydraulic fluids) or decontamination fluids. CSB
management will be responsible for fluid sampling, analysis, and disposal as described above.

Safety Considerations. The liquid waste collection system is classified general service.

A2.8.7 Sanitary Water

The 2-in. sanitary water line is blanked off underground approximately 5 ft north of the
support area building (Figure A2-4). The sanitary water system is designed and limited to
supplying clean water for possible future sanitary use (e.g., toilets, sinks, showers, drinking
fountains) if required. This line connects to the sanitary main 3-in. line located north of the
support area building.

Safety Considerations. The sanitary water system is classified general service.

A2.9 AUXILIARY SYSTEMS

This section includes information on the remaining portions of the CSB that have not been
covered by the preceding sections and that are necessary to provide information on the facility as
it pertains to the hazard and accident analyses.

A2.9.1 Communications System

The design of the plant communications system provides the cabling and/or raceway system
equipment for the telephone, public address, intercom, and radio communications systems within
the CSB to the communications equipment interface point. All communications system physical
interfaces are located in the communications equipment area of the CAEM and UPS room
(Figure A2-3). A door from the electrical equipment room provides access to the CAEM and
UPS room. Space is provided for telephone exchange equipment, paging equipment, wire
terminating frames, radio equipment, and coaxial and fiber-optic transmission equipment. Inside
cable design provides 50% spare cable and will provide spare raceway for inaccessible and/or
embedded areas.
Radio communications within the CSB include hand-held radios, a UHF station and base unit in the control room, an outside antenna mounted on the northeast corner of the operating shelter roof, and an interconnecting cable between the base unit and the antenna. The CSB UHF radio communication system provides for communications between the control room and personnel inside and outside of the operating area shelter and the support area building.

Telephone requirements are based on providing internal and external communications service to all office and operating areas within the CSB. The telephone system provides tie-ins to the Hanford Site crash alarm and the Hanford Site 911 emergency telephone system. An intercom system is incorporated into the telephone system.

Public address system coverage is required for all areas accessible to personnel, both interior and exterior. Public address systems within the CSB are accessed through specified telephone sets within the facility. Master overrides originate from the control room and the emergency response center. An evacuation alarm, complying with DOE Order 6430.1A, Section 1300-6.5.5, will be annunciated through the public address system speakers. The evacuation alarm can be activated from the control room or the emergency response center.

Area alarms are part of the 200 East alarm system, and plant alarms are incorporated into the DCS. Ringdown phones are installed for security use. Fire alarm and detection system signals (discussed in Section A2.7.1) are transmitted to the central fire alarm system by means of a radio system integral to the fire alarm control panel. Radiation and hazardous condition alarms are annunciated by means of the DCS. The Hanford Site crash alarm is annunciated through the general plant telephone system ringer and is directed to selected plant telephones.

**Safety Considerations.** The communications system is classified general service.

### A2.9.2 Security System

All areas of the building have locking Medeco doors and an alarm system. The operating area also has closed-circuit television cameras as part of security and safeguards protection.

Limited area entrances have special access hardware to control unauthorized entry. The large exterior openings have telescoping doors and heavy rolling shield gates (Figure A2-18) to minimize vehicle penetration during the design basis tornado at those locations. The concrete curb around the perimeter of the operations area shelter prevents vehicle penetration for the balance of the building. All vents, shafts, and ducts have intrusion barriers appropriate to their configuration. Limited area entrances have special outside access hardware for intrusion protection. All exit doors are equipped with crash bars for emergency exit.

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3Medeco is a trademark of Medeco, Incorporated.
Security seals are installed between tube plug assemblies and the tube assemblies. This security measure helps to prevent inadvertent removal of a tube plug and also will indicate unauthorized tampering.

Safety Considerations. The security system is classified general service.

A2.9.3 Lightning Protection

The vault intake stack, vault exhaust stack, and ventilation exhaust stack each have lightning protection provided in accordance with NFPA 780. Two exposed cables located on opposite sides of the vault intake and vault exhaust stacks are attached to the stacks about 3 ft above the concrete base. The exposed cables are connected to the support building exhaust stack about 3 ft above roof level and connect to lightning protection ground rods (in the ground test well), which in turn are connected to the building ground grid. The stacks do not need separate air terminals (lightning rods) because they have an electrically continuous metal thickness greater than the 3/16 in. required by NFPA 780 (metal thickness varies for each stack, but no stack has a metal thickness less than 3/8 in.). Air terminals have been installed on the operating shelter roof and the support building roof to provide lightning protection for the buildings. Lightning protection ground conductors also connect these air terminals to the building ground grid. The stacks, air terminals, and ground conductors provide lightning protection for the CSB in accordance with NFPA 780.

Safety Considerations. The lightning protection system is classified general service.

A2.10 REFERENCES


ANSI/ACI 349-85, 1985, Code Requirements for Nuclear Safety Related Concrete Structures, American Concrete Institute, Detroit, Michigan.


Annex A — Canister Storage Building


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Annex A — Canister Storage Building


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Figure A2-1. Canister Storage Building.
Figure A2-2. Multi-Canister Overpack Assembly.
Figure A2-3. Canister Storage Building (sheet 1 of 2)
Figure A2-3. Canister Storage Building. (sheet 2 of 2)
Figure A2-4. Site Plan Showing Water Lines, Grading and Drainage.
Figure A2-5. Canister Storage Building Vault Mat Plan.
Figure A2-6. Canister Storage Building Vault Cross Sections.

- SOUTH - NORTH SECTION
  - INTAKE PLENUMS

- NORTH - SOUTH SECTION
  - EXHAUST PLENUMS

- WEST - EAST SECTION
  - CANISTER STORAGE VAULT 2
  - INTAKE AND EXHAUST PLENUMS
Figure A2-7. Canister Storage Building Vault Operating Floor Plan.
Figure A2-8. Standard and Overpack Tube, Plug, and Embed. (sheet 1 of 2)
Figure A2-8. Standard and Overpack Tube, Plug, and Embed. (sheet 2 of 2)
Figure A2-9. Cask Receiving Pit Load-In/Load-Out Area. (sheet 1 of 8)
Figure A2-9. Cask Receiving Pit Load-In/Load-Out Area. (sheet 2 of 8)
Figure A2-9. Cask Receiving Pit Load-In/Load-Out Area. (sheet 3 of 8)
Figure A2-9. Cask Receiving Pit Load-In/Load-Out Area. (sheet 4 of 8)
Figure A2-9. Cask Receiving Pit Load-In/Load-Out Area. (sheet 5 of 8)
Figure A2-9. Cask Receiving Pit Load-In/Load-Out Area. (sheet 6 of 8)
Figure A2-9. Cask Receiving Pit Load-In/Load-Out Area. (sheet 7 of 8)
Figure A2-9. Cask Receiving Pit Load-In/Load-Out Area. (sheet 8 of 8)
Figure A2-10. Storage Tubes.
Figure A2-11. Tube Base Assembly for Standard and Overpack Storage Tubes. (sheet 1 of 2)
Figure A2-11. Tube Base Assembly for Standard and Overpack Storage Tubes. (sheet 2 of 2)
Figure A2-12. Bellows Assembly for Standard and Overpack Tubes. (sheet 1 of 2)
Figure A2-12. Bellows Assembly for Standard and Overpack Tubes. (sheet 2 of 2)
Figure A2-13. Impact Absorbers for Standard Storage Tubes. (sheet 1 of 2)

TARGET FOR MHM ALIGNMENT

GENERAL ARRANGEMENT FOR LONG AND SHORT TUBE COMBINATIONS

2.13 IN. THICK FLANGE PLATE
0.88 IN. THICK TOP PLATE
3.5 IN. Ø CRUSH TUBE LONG
3.5 IN. Ø CRUSH TUBE SHORT
0.88 IN. THICK INTERMEDIATE PLATE
0.5 IN. THICK BASE PLATE

CABLE ASSEMBLY (TYP. OF 3)

(36.53 PRIOR TO FACTORY CRUSH
34.65) AFTER FACTORY CRUSH

STANDARD BOTTOM IMPACT ABSORBER ASSEMBLY

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Figure A2-13. Impact Absorbers for Standard Storage Tubes. (sheet 2 of 2)

STANDARD INTERMEDIATE IMPACT ABSORBER ASSEMBLY
Figure A2-14. Standard and Overpack Tube Plugs. (sheet 1 of 2)
Figure A2-14. Standard and Overpack Tube Plugs. (sheet 2 of 2)
Figure A2-15. Tube Plug Cover for Standard and Overpack Tubes (Vault 1).

1 - STANDARD TUBE PLUG COVER ASSEMBLY
2 - OVERPACK TUBE PLUG COVER ASSEMBLY
Figure A2-16. Embed Cover for Storage Tube Embeds (Vaults 2 and 3).

1 - STANDARD EMBED COVER ASSEMBLY
2 - OVERPACK EMBED COVER ASSEMBLY
Figure A2-17. Shield Hatch and Multi-Canister Overpack Guide Assembly.
Figure A2-18. Rolling Shield Gates. (sheet 1 of 2)
Figure A2-18. Rolling Shield Gates. (sheet 2 of 2)
Figure A2-19. Canister Storage Building Receiving Crane.

RECEIVING CRANE
(X) LOAD PATH COMPONENTS
Figure A2-20. Cask Transportation System Arrangement Drawing.
Figure A2-21. Cask Lifting Yoke.

CASK LIFTING YOKE

THIS WILL KEEP THE SHIPPING CASK WITHIN THE 60" DROP HEIGHT.
Figure A2-22. Empty Multi-Canister Overpack Transport Dollies.
Figure A2-23. Receiving Crane/Multi-Canister Overpack Handling Machine Anticollision System. (sheet 1 of 4)
Figure A2-23. Receiving Crane/Multi-Canister Overpack Handling Machine Anticollision System. (sheet 2 of 4)
Figure A2-23. Receiving Crane/Multi-Canister Overpack Handling Machine Anticollision System. (sheet 3 of 4)
Figure A2-23. Receiving Crane/Multi-Canister Overpack Handling Machine Anticollision System. (sheet 4 of 4)
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Figure A2-25. Cask Servicing Operations.
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Figure A2-27. Interface Guide Ring Funnel. (sheet 1 of 2)

**STANDARD INTERFACE GUIDE RING FUNNEL ASSEMBLY**
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Figure A2-28. Multi-Canister Overpack Handling Machine Cask Turret System.
Figure A2-29. Rail Stops and Rail Frogs.
Figure A2-30. Multi-Canister Overpack Handling Machine Trolley Seismic X-Restraints.
Figure A2-31. Cask Receiving Pit Operations, Receiving Crane and Tent Gantry Hoist.
Figure A2-32. Cask Receiving Pit Operations, Multi-Canister Overpack Handling Machine.
Figure A2-33. Multi-Canister Overpack Handling Machine Cask Shielding.
Figure A2-34. Turret Locking Pin.
Figure A2-35. Turret Locking Pin Assemblies.
Figure A2-36. Base Locking Pin.
Figure A2-37. Tube Plug Hoist and Grapple.

SCREW JACK LOWERING.
GRAPPLE SEATED ON PLUG.
SCREW JACK UNLOCKS GRAPPLE
AT "POSITION 2" THEN STOPS
AFTER FURTHER TRAVEL OF "Y"

"POSITION 2"

CAM
(NOT LOCKED)

JAW OPEN
(SHOWN AT
FULL EXTENT)

JAW CLOSED

"POSITION 1"

CAM (LATCHED)

SCREW JACK HOISTING.
SCREWJACK JUST CONTACTS
GRAPPLE AT "POSITION 1"
PLUG IS RAISED AFTER
FURTHER TRAVEL OF "X"

TUBE PLUG GRAPPLE
Figure A2-38. Multi-Canister Overpack Grapple. (sheet 1 of 2)
Figure A2-38. Multi-Canister Overpack Grapple. (sheet 2 of 2)
Figure A2-39. Multi-Canister Overpack Handling Machine Ventilation and Compressed Air System. (sheet 1 of 2)
Figure A2-39. Multi-Canister Overpack Handling Machine Ventilation and Compressed Air System. (sheet 2 of 2)
Figure A2-40. Section of Multi-Canister Overpack Sampling/Weld Station Pit. (sheet 1 of 4)
Figure A2-40. Section of Multi-Canister Overpack Sampling/Weld Station Pit. (sheet 2 of 4)
Figure A2-40. Section of Multi-Canister Overpack Sampling/Weld Station Pit. (sheet 3 of 4)
Figure A2-40. Section of Multi-Canister Overpack Sampling/Weld Station Pit. (sheet 4 of 4)
Figure A2-41. Sampling/Weld Station Gantry Crane Section – Sampling Operations.
Figure A2-42. Multi-Canister Overpack Sampling System.
Figure A2-43. Sampling/Weld Station Gantry Crane Floor Plan.
Figure A2-44. Sampling/Weld Station Gantry Crane Section – Weld Operations.
Figure A2-45. Inert Gas Supply Systems.
Figure A2-46. Computer Controlled Gas Tungsten Arc Welding System.
Figure A2-47. Tube Vent and Purge Cart Assembly. (sheet 1 of 2)
Figure A2-47. Tube Vent and Purge Cart Assembly. (sheet 2 of 2)
Figure A2-48. Canister Storage Building Tube Vent and Purge System.
Figure A2-49. Canister Storage Building Confinement Barriers.

- **Trailer Vestibule Operations**
  - MCO Shell
  - Transportation Cask Wall
  - Operating Area Shelter

- **Cask Receiving Pit Operations**
  - MCO Shell
  - Shield Hatch Assembly
  - Operating Area Shelter

- **MHM Operations**
  - MCO Shell
  - MHM MCO Cavity / Vent System
  - Operating Area Shelter

- **Standard Storage Tube Operations**
  - MCO Shell
  - Standard Storage Tube
  - Operating Area Shelter

- **Sampling / Weld Station Operations**
  - MCO Shell
  - Sampling / Weld Station Sample Hood
  - Operating Area Shelter

- **Overpack Storage Tube Operations (Off-Normal)**
  - MCO Shell
  - Overpack Storage Tube / Tube Plug
  - Lock Down Device / Tube Vent and Purge Cart
  - Operating Area Shelter
Figure A2-50. Canister Storage Building Operations - Block Flow Diagram.

**LEGEND**
- CASK/MCO
- CASK/NEW EMPTY MCO
- MECH SEALED MCO
- MONITORED MCO
- OFF-NORMAL MCO
- WELD SEALED MCO

**CANISTER STORAGE BUILDING OPERATIONS BLOCK FLOW DIAGRAM**
Figure A2-51. Heating, Ventilation, and Air Conditioning System Composite Diagram.
Figure A2-52. Stack Monitoring Equipment. (sheet 1 of 2)

ST-989
FRONT VIEW
(Cabinet doors removed for clarity.)
Figure A2-52. Stack Monitoring Equipment. (sheet 2 of 2)
Figure A2-53. Health Protection.

**ITEM - DESCRIPTION**

1. Alpha Continuous Air Monitor
2. Beta Continuous Air Monitor
3. Record Air Sampler
4. Area Radiation Monitor
5. Body Frisker (Typical Location)
6. Hand and Foot Monitor (Typical Location)
7. Alpha/Beta Counter (Typical Location)
8. GE (Germanium) Detector (Typical Location)
9. Multi-Channel Analyzer (Typical Location)
10. CAEM Primary (Continuous Airborne Effluent Monitor)
11. GEMS (Gas Effluent Monitor System)
12. Remote Terminal Unit
13. Accelerometer

*INTENDED LOCATION - Actual location will be based on the Supporting Document for workplace air sampling in the CSB

**INTENDED LOCATION - Actual location will be based on the Supporting Document for area radiation monitors in the CSB

LOCATION OF MONITORING EQUIPMENT
Figure A2-54. Electrical One Line Diagram.
(sheet 1 of 7)
Figure A2-54. Electrical One Line Diagram.
(sheet 2 of 7)
Figure A2-54. Electrical One Line Diagram. (sheet 3 of 7)
MC-33-208 - 480V MOTOR CONTROL CENTER
CANISTER STORAGE BUILDING
Figure A2-54. Electrical One Line Diagram.
(sheet 5 of 7)
Figure A2-54. Electrical One Line Diagram.
(sheet 6 of 7)
TO MC-33-210
480V MCC
ALTERNATE POWER SUPPLY

TO MC-32-207
480V MCC
NORMAL POWER SUPPLY

BA-33-301
UPS BATTERY CABINET

Synchronizing

Maintenance Bypass Switch

Static Transfer Switch with Manual Bypass

V

Neutral

SPARE

UPs Common Trouble Alarm
DES RTU #1

DA-33-304
UPS Distribution Panel "F"
208Y/120V, 3Ph, 4W

DA-33-303
UPS Distribution Panel "C"
208Y/120V, 3Ph, 4W

20AT

UP-33-213
UNINTERRUPTIBLE POWER SUPPLY
CANISTER STORAGE BUILDING
Figure A2-55. Distributed Control System.
Figure A2-56. Compressed Air System.
(sheet 1 of 3)

**CX-1A & CX-1B**

**COMPRESSOR**

DESIGN CAP: 200 CFM @ 125 PSIG DISCHARGE PRESSURE
HORSEPOWER: 60 HP
FAN: 5 HP

**SEQUENCE CONTROL UNIT**

**INTAKE AIR**

**TO ATM**

**AIR FILTER**

**BLOWDOWN**

**F**

**TO ATM**

**F**

**GEAR CASE**

**1ST STAGE**

**2ND STAGE**

**COMPRESSED AIR**

**HYDRAULIC FLUID**

**COMPRESSED AIR SYSTEM**

**AIR COMPRESSORS**

**COMPRESSOR COMMON PUMP**

**CONDENSATE**

**TO COMPRESSED AIR SYSTEM**

**AIR COMPRESSORS**

**AF2-87 March 2000**
Figure A2-56. Compressed Air System.  (sheet 2 of 3)

**F-5**  
**PREFILTER**  
DESIGN CAP: 200 SCFM  
DESIGN PRES: 150 PSIG  
DESIGN TEMP: 150°F  
LIMIT ΔP: 0-60 PSI

**DR-1**  
**AIR DRYER**  
DESIGN CAP: 200 SCFM  
DESIGN PRES: 150 PSIG  
DESIGN TEMP: 250°F

**F-6**  
**AFTERFILTER**  
DESIGN CAP: 200 SCFM  
DESIGN PRES: 150 PSIG  
DESIGN TEMP: 150°F  
LIMIT ΔP: 0-60 PSI

---

**COMPRESSED AIR SYSTEM**  
**AIR DRYER**

---

March 2000
Figure A2-56. Compressed Air System.
(sheet 3 of 3)
Figure A2-57. Liquid Waste Collection System. (sheet 1 of 2)

**SU-1**
WASTE WATER SUMP
CONCRETE W/EPOXY LINING
4' x 3' x 3' DEEP
~ 110 GALLONS

**P-4**
WASTE WATER SUMP PUMP
DESIGN CAPACITY: 10 GPM
DESIGN TEMPERATURE: 100°F
DESIGN PRESSURE: 5 PSIG
AIR POWERED PORTABLE

**55 GAL DRUM**

**SUCTION STRAINER**
FOOT VALVE

**LIQUID WASTE COLLECTION SYSTEM**
**SUMP 1**
Figure A2-57. Liquid Waste Collection System. (sheet 2 of 2)

SU-2
WASTE WATER SUMP
CONCRETE W/EPoxy LINING
2'x2'x2' DEEP
~ 55 GALLONS
This document was too large to scan as a single document. It has been divided into smaller sections.

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CHAPTER A3.0

HAZARD AND ACCIDENT ANALYSES
HNF-3553 REV 0
Annex A — Canister Storage Building

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LIST OF TERMS

ALARA as low as reasonably achievable
ARF airborne release fraction
ARR airborne release rate
BDBA beyond design basis accident
BR breathing rate
CSB Canister Storage Building
CVDF Cold Vacuum Drying Facility
DBA design basis accident
DBE design basis earthquake
DOE U.S. Department of Energy
EF error factor
FFTF Fast Flux Test Facility
FSAR final safety analysis report
HEPA high-efficiency particulate air (filter)
HVAC heating, ventilation, and air conditioning
ITS important to safety
LPF leak path factor
MAR material at risk
MCO multi-canister overpack
MHM multi-canister overpack handling machine
MTU metric ton of uranium
NA not applicable
NPH natural phenomena hazard
NRC U.S. Nuclear Regulatory Commission
PMP probable maximum precipitation
RF respirable fraction
SNF spent nuclear fuel
SNV standard normal variable
SSC structure, system, and component
TSR technical safety requirement
UD unit dose
A3.0 HAZARD AND ACCIDENT ANALYSES

A3.1 INTRODUCTION

This chapter presents a summary of the key methodology, assumptions, and results of the final safety hazard analysis and design basis accident (DBA) analyses that have been performed for the final design and operation of the Canister Storage Building (CSB). These analyses form a Safety Basis for the final safety analysis report (FSAR) and present a comprehensive evaluation of the CSB handling- and storage-related activities and natural phenomena and external hazards that can affect the public, workers, and environment. Single and multiple initiating events from equipment and human error failures in the facility, and human and natural events (i.e., common mode failure) outside of the facility have been considered. When the FSAR is approved by the U.S. Department of Energy (DOE), this Safety Basis will establish an Authorization Basis for the Spent Nuclear Fuel (SNF) Project. Changes to the facility during operations will be reviewed to ensure they do not affect the Authorization Basis and if necessary new analyses will be performed. This review process is described in Chapter 17.0 of the SNF Project FSAR and is termed the unreviewed safety question process.

This chapter has been prepared following the direction provided in Chapter 3.0 of the SNF Project FSAR. Where appropriate, this chapter references Chapter 3.0 of the SNF Project FSAR for additional details regarding the accident analysis evaluation guidelines and limits and analysis methodology. The contents of this chapter are as follows:

- The requirements for establishing the Safety Basis for the CSB are listed in Section A3.2. The requirements listed consist of DOE orders and standards and applicable U.S. Nuclear Regulatory Commission (NRC) rules and guidance.

- The CSB hazard analysis is summarized in Section A3.3. The complete hazard analysis is contained in HNF-SD-SNF-HIE-001, Canister Storage Building Hazard Analysis Report. The analysis identifies hazard sources, hazardous conditions, potential accident scenarios and their initiators, and assessments of event frequencies and consequences. Hazards are identified by form and location and represent a complete spectrum of events that could occur throughout the facility. An initial set of safety features that would serve to prevent or mitigate the postulated accident scenarios are identified in the hazard analysis, with a final set of safety features identified in the accident analyses in Section A3.4.2 and in Chapters A4.0 and A5.0. A safety feature that would prevent an accident scenario is defined as one that reduces the expected annual frequency of occurrence for the accident to beyond extremely unlikely (less than $10^{-6}$ per year). Hazard analysis methodology and evaluation criteria are discussed in further detail in Section 3.3 of the SNF Project FSAR.
The final facility hazard classification, determined in accordance with DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*, is addressed in Section A3.3.2.2. The CSB has been assigned a final designation of hazard category 2 facility based on material at risk (MAR).

Section A3.3.2.3 contains discussions of defense in depth, worker safety, and environmental protection, including a detailed tabulation of engineered and administrative features that have been identified as providing for worker safety.

The hazard evaluation ranking performed in the hazard analysis identifies hazards and associated events that pose a challenge to offsite and onsite radiological dose evaluation guidelines. This ranking serves as the basis for selecting unique and representative accidents for further detailed quantitative evaluation as DBAs. These are presented in Section A3.4.2. Each of the DBAs analyzed represents a bounding case for a category of hazards and accidents. SNF-4042, *Evaluation of Accident Frequencies at the Canister Storage Building*, documents the derivation of the DBA frequency range. The six major DBA categories are evaluated in the following sections:

- Section A3.4.2.1, "Mechanical Damage of Multi-Canister Overpack"
- Section A3.4.2.2, "Gaseous Release from the Multi-Canister Overpack"
- Section A3.4.2.3, "Multi-Canister Overpack Internal Hydrogen Deflagration"
- Section A3.4.2.4, "Multi-Canister Overpack External Hydrogen Deflagration"
- Section A3.4.2.5, "Thermal Runaway Reactions inside the Multi-Canister Overpack"
- Section A3.4.2.6, "Violations of Design Temperature Criteria."

Three receptor locations were used for the DBA analysis:

- Hanford Site boundary (17,390 m east of the CSB) — defined release limits; used for calculation of offsite doses and selection of safety-class features (Sellers 1997, Scott 1995)
- Onsite worker (100 m east from the CSB) — defined risk evaluation guidelines; used for calculation of onsite doses and selection of safety-significant features (Sellers 1997)
Highway 240 (onsite, approximately 9,280 m west of the CSB) — no defined evaluation guideline; doses calculated for informational purposes only (Scott 1995).

For the analyzed DBAs, safety-class and safety-significant features have been selected from the candidate features identified in the hazard analysis. When required by the SNF Project’s commitments to meet equivalent NRC requirements, candidate features have been identified as “important to safety” and designed, engineered, and procured consistent with safety-class or safety-significant classification requirements. Safety structures, systems, and components (SSCs) and controls are presented first in Section A3.4.2 with the discussion of each accident and described in more detail in Chapters A4.0 and A5.0.

Each of the six DBAs that have been quantitatively analyzed represents a bounding case for a category of hazards and accidents. An in-depth review was performed of all remaining significant accidents identified by the hazard analysis in each of the categories. The accidents reviewed typically represented the same accidents with different initiators, or a different accident bounded by the case presented. This review resulted in the selection of additional safety-class and safety-significant SSCs and technical safety requirements (TSRs) to ensure that all individual hazards and accidents within a category have been addressed. The table and text that accompany each DBA in Section A3.4.2 include the preventive and mitigative features and associated TSRs for the bounding case presented and for all other events within the accident category. Defense-in-depth features also are identified in these tables.

This chapter interfaces with several other SNF Project safety documents. This chapter identifies and evaluates material hazards associated with the CSB and analyzes bounding accident scenarios associated with multi-canister overpack (MCO) handling and storage activities that occur within the CSB. The planned scope and content of various other SNF Project safety reports, TSRs, and supporting safety documents are defined in HNF-SD-SNF-PLN-012, Spent Nuclear Fuel Project Integrated Safety Management Plan. The hazards associated with transport and accident scenarios that are postulated during shipment are analyzed in HNF-SD-TP-SARP-017, Safety Analysis Report for Packaging, Onsite, Multi-Canister Overpack Cask.

To protect the limiting values assumed in the hazard and DBA analyses, the interface between the Cold Vacuum Drying Facility (CVDF) and the CSB requires that an MCO delivered to the CSB contain less than 200 g of free water and provide confinement of the combustible hydrogen gases within the mechanically sealed MCOs. In addition, the CSB hazard and accident analysis relies on the SNF in an MCO to be clean (i.e., loose surface contaminants removed) and the fuel loading to be controlled (e.g., mass of fuel in a scrap basket, number of scrap baskets). The principal assumptions associated with K Basins and CVDF performance bounding limits are taken from HNF-SD-SNF-TI-059, A Discussion on the Methodology for Calculating Radiological and Toxicological Consequences for the Spent Nuclear Fuel Project at the Hanford Site, and they are identified in Section A3.4.2. Safety-related performance documentation from the K Basins and the CVDF is relied on to ensure that the as-received content and condition of...
the MCO are as required for the safety basis properties. The use of overpack tubes is related to recovery actions of short duration. The SNF Project will have direction in operating procedures that states the use of the overpack tubes is for recovery operations for potentially damaged MCOs or MCOs whose status or configuration is uncertain. A recovery team will convene in accordance with operational procedures. This team will determine recovery actions, including the conditions for use of overpack tubes. Actions using the overpack tubes will be for a short-term period, less than one year. Use of overpack tubes for other than recovery actions will be subject to the unreviewed safety question process.

In this chapter, there are references to the condition of the fuel in the MCO or the MCOs in the CSB facility as "safe and stable." In this context a "stable" condition is defined as either (1) an energy-balanced steady-state condition within temperature and pressure limits, or (2) a transient operating condition for which administrative controls are implemented to ensure the system can be placed into an energy-balanced steady-state condition before temperature or pressure limits are exceeded. A "safe" condition is defined as one in which one or more confinement barriers exist and are operative. This definition is consistent with HNF-SD-SNF-SARR-005, Multi-Canister Overpack Topical Report, Chapter 4.0.

A3.2 REQUIREMENTS

Chapter 3.0 of the SNF Project FSAR lists the design codes, standards, regulations, and DOE orders that contain requirements and guidance for establishing the safety basis for the SNF Project. The requirements for the CSB that pertain to the safety analysis are provided here.

- DOE Order 5480.23, Nuclear Safety Analysis Reports, in conjunction with its Attachment 1, "Interim Guidance for DOE Order 5480.23," sets the requirements for analysis. This chapter complies with these requirements by documenting performance of hazard and accident analyses. The methodology, assumptions, and criteria used to identify facility hazards, hazard rankings, candidate accidents, DBAs, preventive and mitigative features and controls and the classification of these features (along with the definition of safety functions, performance criteria, and applicability) are described in Chapter 3.0 of the SNF Project FSAR and documented in this chapter.

- DOE Order 5480.22, Technical Safety Requirements. This order sets the requirements for the development and preparation of a TSR document. This chapter complies with these requirements by documenting the performance of hazard and accident analyses in accordance with HNF-PRO-704, Hazard and Accident Analysis Process, and DOE Order 5480.23. The results of the analyses were used to identify specific SSC safety functions, performance requirements for the SSCs, and the times for application of the safety functions.
DOE Order 6430.1A, General Design Criteria. This order provides requirements for the identification of safety-class items. The analyses documented in this chapter used the SSC classification requirements of DOE Order 6430.1A in the identification of safety-function SSCs and their safety classification.


In Letter 95-SFD-167, Implementation of K Basins Spent Nuclear Fuel Project (SNFP) Regulatory Policy (Sellers 1995), DOE established the requirement for new SNF Project facilities to achieve “nuclear safety equivalency” to comparable NRC-licensed facilities. The SNF Project identified the NRC requirements that were to be met in addition to existing and applicable DOE requirements to establish nuclear safety equivalency. These NRC requirements and the process used to identify them are documented in HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements and in WHC-SD-SNF-DB-009, Canister Storage Building Natural Phenomena Hazards. These documents established the design requirements to be met for the CSB to achieve NRC equivalency and considered the following:

Title 10, Code of Federal Regulations, Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste” (10 CFR 72). This rule is used for the licensing of independent spent fuel storage installations. Section 72.122, “Overall Requirements,” requires that the design bases for SSCs important to safety reflect appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. Section 72.24, “Content of Application: Technical Information,” provides requirements in Paragraph 72.24(m) for the analyses of accidents and natural phenomena events that could result in a dose at the controlled area boundary.

NRC Regulatory Guide 3.48, Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage). This guide establishes the format and content for safety analysis reports for license applications for fuel storage facilities.

While these documents have particular significance to this chapter, they do not, by themselves, establish requirements for the CSB or the Chapter A3.0 accident analyses. Requirements identified in HNF-SD-SNF-DB-003 and WHC-SD-SNF-DB-009 address NRC safety equivalency. These requirements include safety documentation for (1) determining “important-to-safety” classifications for preventive and mitigative features; (2) identifying and resolving worker safety issues in DOE Order 6430.1A, DOE Order 5480.23, and DOE-STD-3009-94; and (3) evaluating nearby activities identified in Section A1.6, “External Human-Generated Threats,” that may represent a threat to the facility.
Important-to-safety SSCs have been identified in accordance with 10 CFR 72, Section 72.3, “Definitions.” Once SSCs have been identified as having a function meeting the definition of important to safety, the requirements for SSCs important to safety specified in 10 CFR 72 are imposed. A graded approach is applied to an SSC important to safety by using the guidance provided in NUREG/CR-6407, Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety, as modified by Letter 98-SFD-172, Contract No. DE-AC06-96RL13200 — Approval of Spent Nuclear Fuel Project Path Forward, Additional Nuclear Regulatory Commission Requirements, HNF-SD-SNF-DB-003 (Sellers 1998):

- **Category A — Critical to Safe Operation**

  SSCs in this category include those whose failure or malfunction could directly result in a condition adverse to public health and safety. Important-to-safety SSCs in this category have been classified as safety class using the definition in DOE Order 6430.1A, with the additional requirements therein.

- **Category B — Major Impact on Safety**

  SSCs in this category include those whose failure or malfunction could result in a condition adversely affecting collocated worker health and safety. Note that from the definition of Category C, Category B is understood to include events that could significantly damage the MCO without severe impact to public health and safety. SSCs in this category are classified as safety significant.

- **Category C — Minor Impact on Safety**

  SSCs whose failure or malfunction would not significantly reduce confinement and would not be likely to create a situation adversely affecting public or collocated workers' health and safety. SSCs in this category are classified as non-safety related and may be credited as defense in depth.

The determination of important to safety is directly incorporated into the tables of identified preventive or mitigative safety function features at the end of each DBA scenario in Section A3.4.2, and external human-generated threats are discussed in Section A3.3.2.3. Requirements of WHC-SD-SNF-DB-009 are incorporated into natural phenomena hazard design criteria and considered in the hazard evaluation. Additional requirements for the MCO are identified in HNF-SD-SNF-SARR-005.

Letter 97-SFD-172, Risk Evaluation Guidelines (REGs) to Ensure Inherently Safer Designs (Sellers 1997), provides the onsite risk evaluation guidelines and offsite release limits that have been used in this chapter for comparison with DBA consequences.
A3.3 HAZARD ANALYSIS

The hazard identification and evaluation process provides a thorough, predominantly qualitative evaluation of the spectrum of risks to the public, workers, and the environment caused by accidents that involve the hazards identified in the hazard analysis. The hazard evaluation was a structured and systematic examination of the CSB facility and its operations using standard industry (American Institute of Chemical Engineers) hazard evaluation techniques. This process is further described in Section 3.3 of the SNF Project FSAR. An initial hazard analysis was prepared to support HNF-SD-SNF-RPT-004, Canister Storage Building Safety Analysis Report — Phase 3: Safety Analysis Documentation Supporting Building Construction, and to integrate all previous CSB hazard analyses. A final hazard analysis has been performed to support the accident analysis and is documented in HNF-SD-SNF-HIE-001. The final hazard analysis systematically reviewed the final CSB design, as described in Chapter A2.0, associated supporting design documentation, and design references to identify any additional hazardous materials or energy sources that have the potential to initiate an accident that could require further review or analysis. This process resulted in the selection of six candidate accidents for more comprehensive analysis in subsections of Section A3.4.2.

The hazard analysis (HNF-SD-SNF-HIE-001) was based on the design and operations described in Chapters A2.0 and A4.0. The analysis included review of the CSB design and reviews of operation flow diagrams and operating procedures. The following normal CSB operations were considered:

- Receiving the transporter containing the cask–MCO and moving it into the facility
- Moving the cask–MCO to the load-in/load-out area and removing the cask lid at the cask receiving pit
- Transporting the MCO from the load-in/load-out area to the storage tube or to the MCO sampling/weld station with the MCO handling machine (MHM)
- Transporting the MCO from the storage tube to the MCO sampling/weld station and returning it to the storage tube after sampling
- Conducting activities during sampling and welding
- Conducting activities during MCO staging and interim storage.

The following off-normal MCO storage operations also were considered in the hazard analysis.

- Completing recovery actions after the event or accident leading to MCO damage has been terminated.
- Placing the off-normal MCO in the overpack storage tube.
- Installing the overpack storage tube plug cover.
- Establishing an inert atmosphere in the overpack tube.

The following key sources of information were used during the hazard analysis to evaluate the hazards (HNF-SD-SNF-HIE-001):


- HNF-SD-TP-SARP-017, *Safety Analysis Report for Packaging, Onsite, Multi-Canister Overpack Cask*, for coverage of accidents involving the transporter and transportation cask and for definition of assumptions inherent in defining the transportation window

- HNF-SD-SNF-SARR-005, *Multi-Canister Overpack Topical Report*, for criteria and assumptions related to the MCO design

- Representatives from the design authority and from facility operations for details of design, operating modes, and procedures

**A3.3.1 Methodology**

The methodology used to identify and evaluate the SNF Project facility hazards is described in detail in Chapter 3.0, Section 3.3, of the SNF Project FSAR. This section discusses those areas of the methodology that are specific to the CSB. The hazard evaluation process identified hazardous conditions, determined causes and preventive and mitigative features, and qualitatively estimated the consequences and frequencies of occurrence. The results of the application of this methodology to the CSB are presented in Section A3.3.2. The hazard analysis was performed in accordance with DOE-STD-3009-94 and implements the requirements of DOE Order 5480.23.
A3.3.1.1 Hazard Identification. The CSB hazard analysis addressed normal CSB operations for handling and storing a sealed MCO (HNF-SD-SNF-HIE-001). In order to assist in unreviewed safety question determination for any future recovery plans, the hazard analysis also identified and analyzed potential hazards associated with storing an off-normal MCO in an overpack storage tube. The hazard analysis evaluated scenarios created by failures of SSCs, human errors, or other accidents leading to potential releases of hazardous material in the facility or other impacts on safety. The hazards associated with transport and the accident scenarios that are postulated to occur during shipment are analyzed in HNF-SD-TP-SARI'-017.

Hazard identification for the CSB was based on examination of seven major areas in the facility:

1. Trailer vestibule (TV)
2. Load-in/load-out area (formerly known as the service area) (SA)
3. Operating area (OA)
4. Sampling/weld station (formerly known as the weld station) (WS)
5. Vault (VL)
6. Support building (SB)
7. Outside (OU).

The hazard analysis included hazards in each area and the process through all areas. If a hazard affected another area, the hazard analysis evaluation either evaluated it in that area or evaluated it in the affected area.

The cask handling, MCO handling, and MCO storage activities that can take place within each area at the CSB were identified and the hazards identified by form (e.g., electrical, thermal, friction), type (e.g., motors, power tools, wiring), and building location. A standardized hazardous material/energy source checklist (see Section 3.3.1.1, Table 3-1 of the SNF Project FSAR) was used to group the potentially hazardous materials and energy sources in each of the seven major facility areas. The two-letter abbreviations shown for the seven areas were used as the checklist location designators in the hazard analysis. The methodology of the hazards identification process is described further in Section 3.3.1.1 of the SNF Project FSAR.

A3.3.1.2 Hazard Evaluation. As described in Section A3.3.1.1, hazards associated with abnormal, accident, or recovery conditions at the CSB were identified. The hazards identified were evaluated to determine the causes of the hazard, potential accidents that could result from the presence of each hazard, and consequences to the public offsite, the collocated and facility workers onsite, the environment, or the CSB. Safety features, segregated into preventive and mitigative features, were identified for each hazard based on the ability of the feature to prevent or mitigate the consequences. Qualitative estimates of the frequency and consequences of the hazardous condition also were assigned (see Section 3.3.1.2 of the SNF Project FSAR for the criteria used in assigning the consequence and frequency categories).
A3.3.2 Hazard Analysis Results

The hazard analysis process is described in HNF-SD-SNF-HIE-001 and summarized in Section 3.3.1 of the SNF Project FSAR and Section A3.3.1. The results of that process, in the order of progression, are as follows:

- A series of checklist-style tables describing hazardous materials and energy sources, organized by major areas of the CSB — These tables were used to develop hazard analysis accident scenarios.

- A series of tables describing standard industrial hazards considered, organized by major area of the CSB — These events were judged to have no contribution to uncontrolled radiological and/or hazardous materials releases and were not considered in the selection of DBAs, safety-class or safety-significant features, or TSRs. They were among the hazards considered and, therefore, are included for completeness.

- A series of tables describing potential hazard scenarios, organized by major area of the CSB — These tables included hazardous energy sources and materials, hazardous conditions, causes and initiators, potential accidents, qualitative determinations of event frequencies and consequences, safety features for prevention and/or mitigation of the consequences, and defense-in-depth or worker safety features.

- A table, organized by major building area of the CSB, assigning risk bins to causes associated with significant consequences to offsite and onsite receptors — Consistent with DOE-STD-3009-94, the events located in risk bins representing "situations of concern," or "situations of major concern," were evaluated as candidate DBAs.

- A final list of candidate DBAs sorted by risk ranking and energy change or release — This list formed the basis for selection of the DBAs presented in Section A3.4.2.

DBA selection is addressed in Section A3.3.2.3.5 and its accompanying Table A3-4. In terms of the risk binning process, the accidents chosen from the hazard analysis for further analysis as DBAs were all events identified in consequence categories S3 and S2. These categories indicate significant effects to offsite and onsite receptors. Some events were brought forward into Table A3-4 that were identified as S1 consequence category events in the hazard analysis. These events were brought into Table A3-4 because they were initially of particular concern and were to be evaluated in more detail. These events are represented in Table A3-4 with a dash (-) in the risk ranking column.

The final hazard analysis has been reconciled with the up-to-date CSB facility design described in Chapters A2.0, A4.0, and A5.0.
A3.3.2.1 Hazard Identification. The final CSB hazard analysis tables are shown in HNF-SD-SNF-HIE-001. The hazards are identified by form (e.g., electrical, thermal, friction), type (e.g., motors, power tools, wiring), and building location (e.g., trailer vestibule, load-in/load-out area, operating area, sampling/weld station, vault, support building). A checklist designator (originating in the CSB hazard analysis tables) associated with each hazard contains a two-letter facility location designation, a letter corresponding to the general type of hazardous energy or material source, and a final number (sometimes with an associated letter) that identifies the specific source of the hazardous energy or material source. The main inventory of hazardous material in the CSB is the radionuclide content of the MCOs. The toxicological hazards of the radionuclide inventory were reviewed. As described in Section 3.4.1.1 of the SNF Project FSAR, the radiological guidelines were found to be more limiting than the toxicological guidelines for the release of SNF particulate. Other hazardous material identified by the hazard identification process includes pyrophoric metals and hydrides, oxidizers, hydrogen, diesel fuel, and other flammable or combustible materials (see Table A3-1). As discussed in HNF-SD-SNF-HIE-001, no hazardous or toxicological materials were identified that are not routinely encountered by the public, thus no unique or specialized institutional controls are required. A specific and comprehensive analysis of all fire hazards associated with the CSB was completed to augment the standard hazard analysis (HNF-SD-SNF-FHA-002).

The CSB does not have an operating history, so major hazards resulting from facility operation cannot be identified or summarized as suggested by DOE-STD-3009-94. However, as described in Section 3.3.2.1 of the SNF Project FSAR, the CSB spent fuel handling and storage activities are similar to those used by the independent spent fuel storage installations that have been issued materials licenses under 10 CFR 72. Major hazards from similar independent spent fuel storage installations were considered when performing the hazard analysis for the CSB. These hazards include generation of combustible gases, failure of confinement barriers, defects in cask integrity, and the spread of external contamination. This was described in the preliminary safety evaluation and the preliminary hazard analysis.

A3.3.2.2 Hazard Classification. A final hazards categorization of the CSB facility was performed based on the final hazard analysis (HNF-SD-SNF-HIE-001) and on accident analysis documentation for the facility provided in SNF-3328, Canister Storage Building Design Basis Accident Analysis Documentation. Consistent with DOE-STD-1027-92, the final categorization was based on the unmitigated releases of the MAR, presented in Table A3-2, from the worst-case beyond design basis accident (BDBA) in Section A3.4.3.4.2. These CSB material quantities were compared against the DOE-STD-1027-92 threshold quantities. The CSB facility final hazard categorization found the CSB facility to be a hazard category 2 facility. This categorization level is consistent with the bases and guidance described in DOE-STD-1027-92.

The established safety basis radiological nuclide inventory per metric ton of uranium (HNF-SD-SNF-TI-015) was used to estimate the material quantities available for release by multiplying the MAR and the specific nuclide inventories per metric ton of uranium (see Table A3-2). These quantities were compared against the category 2 threshold values from DOE-STD-1027-92, Table A.1 (see Table A3-2). The inventories in Table A3-2 are based on 100 kg of UO$_2$, which is derived from the worst-case BDBA consequence in Section A3.4.3.1.
<table>
<thead>
<tr>
<th>Field name or location</th>
<th>MAR-subject</th>
<th>MAR-description</th>
<th>MAR-classification</th>
<th>Capacity</th>
<th>Material type</th>
<th>Physical form</th>
<th>Volume or activity</th>
<th>Transient</th>
<th>Quantity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Truck vestibule</td>
<td>MCO in cask</td>
<td>SNF and particulate matter in MCO</td>
<td>SNF in MCO</td>
<td>1,000 L per MCO; two loaded scrap baskets and four loaded Mark IA fuel baskets MCO estimated to contain up to 34 kg of particulate matter after cold vacuum drying and shipping to CSB</td>
<td>Mark IA or Mark IV spent fuel from N Reactor</td>
<td>Solid consisting of fuel and particulate corrosion products</td>
<td>Slightly greater than 6 MTU at 4.11 E+04 Ci/MTU</td>
<td>Yes, particulate is transient</td>
<td>Slightly greater than 6 MTU per MCO including up to 34 kg particulate per MCO</td>
</tr>
<tr>
<td>Hydrogen gas</td>
<td>Combustible gas</td>
<td>&lt;1 m³ of hydrogen gas per MCO</td>
<td>Hydrogen gas</td>
<td>Gas</td>
<td>&lt;1 m³ of hydrogen gas per MCO</td>
<td>Yes</td>
<td>&lt;1 m³ of hydrogen gas per MCO</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Truck fuel tank</td>
<td>Diesel fuel</td>
<td>Diesel fuel</td>
<td>100 gal</td>
<td>NA</td>
<td>Liquid</td>
<td>Up to 100 gal</td>
<td>Yes</td>
<td>100 gal</td>
<td></td>
</tr>
<tr>
<td>Load-in' load-out area</td>
<td>MCO lifted from cask</td>
<td>SNF and particulate matter in MCO</td>
<td>Finely divided particulate matter</td>
<td>MCO estimated to contain up to 34 kg of particulate matter after cold vacuum drying and shipping to CSB</td>
<td>NA</td>
<td>Solid</td>
<td>Up to 34 kg at 4.11 E+04 Ci/MTU</td>
<td>Yes, particulate is transient</td>
<td>Up to 34 kg per MCO</td>
</tr>
<tr>
<td>Hydrogen gas</td>
<td>Combustible gas</td>
<td>&lt;1 m³ of hydrogen gas per MCO</td>
<td>Hydrogen gas</td>
<td>Gas</td>
<td>&lt;1 m³ of hydrogen gas per MCO</td>
<td>Yes</td>
<td>&lt;1 m³ of hydrogen gas per MCO</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Operating area</td>
<td>MCO contents</td>
<td>SNF and particulate matter</td>
<td>Finely divided particulate matter</td>
<td>Two loaded scrap baskets and four loaded Mark IA fuel baskets MCO estimated to contain up to 34 kg of dispersible particulate matter after cold vacuum drying, shipping to CSB, and sampling</td>
<td>Mark IA or Mark IV spent fuel from N Reactor</td>
<td>Solid consisting of fuel and particulate corrosion products</td>
<td>Slightly greater than 6 MTU at 4.11 E+04 Ci/MTU</td>
<td>Yes, particulate is transient</td>
<td>Slightly greater than 6 MTU including up to 34 kg particulate per MCO</td>
</tr>
<tr>
<td>Hydrogen gas</td>
<td>Combustible gas</td>
<td>&lt;1 m³ of hydrogen gas per MCO</td>
<td>Hydrogen gas</td>
<td>Gas</td>
<td>&lt;1 m³ of hydrogen gas per MCO</td>
<td>Yes</td>
<td>&lt;1 m³ of hydrogen gas per MCO</td>
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<td></td>
</tr>
<tr>
<td>Contents of a loaded storage tube</td>
<td>SNF and particulate matter in two MCOs</td>
<td>NA</td>
<td>Two MCOs; each loaded with two scrap baskets and four Mark IA fuel baskets</td>
<td>Mark IA or Mark IV spent fuel from N Reactor</td>
<td>Solid consisting of fuel and particulate corrosion products</td>
<td>Slightly greater than 12 MTU at 4.11 E+04 Ci/MTU</td>
<td>Yes, particulate is transient</td>
<td>Slightly greater than 12 MTU including 68 kg particulate per storage tube</td>
<td></td>
</tr>
<tr>
<td>Hydrogen gas</td>
<td>Combustible gas</td>
<td>-2 m³ of hydrogen gas per storage tube</td>
<td>Hydrogen gas</td>
<td>Gas</td>
<td>-2 m³ of hydrogen gas per storage tube</td>
<td>Yes</td>
<td>-2 m³ of hydrogen gas per storage tube</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Field name or location</td>
<td>MAR-subject</td>
<td>MAR-description</td>
<td>MAR-classification</td>
<td>Capacity</td>
<td>Material type</td>
<td>Physical form</td>
<td>Volume or activity</td>
<td>Transient</td>
<td>Quantity</td>
</tr>
<tr>
<td>------------------------</td>
<td>-------------</td>
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<td>-------------------</td>
<td>----------</td>
<td>---------------</td>
<td>---------------</td>
<td>-------------------</td>
<td>-----------</td>
<td>----------</td>
</tr>
<tr>
<td>Sampling/weld station</td>
<td>MCO in sampling/weld station</td>
<td>SNF and particulate matter in MCO</td>
<td>SNF in MCO</td>
<td>Two loaded scrap baskets and four loaded Mark IA fuel baskets</td>
<td>Mark IA or Mark IV spent fuel from Reactor</td>
<td>Solid consisting of fuel and particulate corrosion products</td>
<td>Slightly greater than 6 MTU at 4.11 E+04 Ci/MTU</td>
<td>Yes, particulate is transient</td>
<td>Slightly greater than 6 MTU per MCO including up to 34 kg particulate per MCO</td>
</tr>
<tr>
<td>Hydrogen gas</td>
<td>Combustible gas</td>
<td>&lt;1 m³ of hydrogen gas per MCO</td>
<td>Hydrogen gas</td>
<td>Gas</td>
<td>&lt;1 m³ of hydrogen gas per MCO</td>
<td>Yes</td>
<td>&lt;1 m³ of hydrogen gas per MCO</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Vault</td>
<td>Contents of all loaded MCOs in vault 1</td>
<td>SNF and particulate matter in MCO</td>
<td>NA</td>
<td>220 storage tubes each loaded with two MCOs</td>
<td>Various types of SNF from Reactor</td>
<td>Solid consisting of fuel and particulate corrosion products</td>
<td>4.11 E+04 Ci/MTU × vault contents</td>
<td>Yes, particulate is transient</td>
<td>Slightly greater than 6 MTU per MCO including up to 34 kg particulate per MCO</td>
</tr>
<tr>
<td>Support building</td>
<td>Particulate on HVAC filters</td>
<td>Radioactive particulate matter from MCOs</td>
<td>Radioactive particulate matter</td>
<td>Minimal</td>
<td>NA</td>
<td>Solid</td>
<td>NA</td>
<td>Yes, particulate is transient</td>
<td>Minimal</td>
</tr>
<tr>
<td>Outside area</td>
<td>Inert gas storage</td>
<td>Helium used to inert MCOs and overpack storage tubes</td>
<td>Helium</td>
<td>12 bottles</td>
<td>Helium gas</td>
<td>Gas</td>
<td>NA</td>
<td>Yes</td>
<td>12 bottles or a gas transport trailer</td>
</tr>
</tbody>
</table>

CSB = Canister Storage Building.
HVAC = heating, ventilation, and air conditioning.
MAR = material at risk.
MCO = multi-canister overpack.
MTU = metric ton of uranium.
NA = not applicable.
SNF = spent nuclear fuel.
Table A3-2. Radionuclide Inventory for the Canister Storage Building Material at Risk Compared with the Category 2 Threshold Quantities. (3 sheets)

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Fuel activity (Ci/MTU)</th>
<th>Material at riska (100 kg or 0.1 MTU)</th>
<th>Category 2 threshold quantitiesb</th>
<th>Ratio of material to category 2 threshold quantity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fission and activation products</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>H-3</td>
<td>2.61 E+01</td>
<td>2.6 E+00</td>
<td>3.00 E+05</td>
<td>8.70 E-06</td>
</tr>
<tr>
<td>C-14</td>
<td>5.53 E-01</td>
<td>5.5 E-02</td>
<td>1.40 E+06</td>
<td>3.95 E-08</td>
</tr>
<tr>
<td>Fe-55</td>
<td>5.41 E-01</td>
<td>5.4 E-02</td>
<td>1.10 E+07</td>
<td>4.92 E-09</td>
</tr>
<tr>
<td>Co-60</td>
<td>2.09 E+00</td>
<td>2.1 E-01</td>
<td>1.90 E+05</td>
<td>1.10 E-06</td>
</tr>
<tr>
<td>Ni-59</td>
<td>3.18 E-02</td>
<td>3.2 E-03</td>
<td>4.30 E+05</td>
<td>7.40 E-09</td>
</tr>
<tr>
<td>Ni-63</td>
<td>3.47 E+00</td>
<td>3.5 E-01</td>
<td>4.50 E+06</td>
<td>7.71 E-08</td>
</tr>
<tr>
<td>Se-79</td>
<td>6.54 E-02</td>
<td>6.5 E-03</td>
<td>4.30 E+05</td>
<td>1.52 E-08</td>
</tr>
<tr>
<td>Kr-85</td>
<td>3.70 E+02</td>
<td>3.7 E+01</td>
<td>2.80 E+07</td>
<td>1.32 E-06</td>
</tr>
<tr>
<td>Sr-90</td>
<td>6.93 E+03</td>
<td>6.9 E+02</td>
<td>2.20 E+04</td>
<td>3.15 E-02</td>
</tr>
<tr>
<td>Y-90</td>
<td>6.93 E+03</td>
<td>6.9 E+02</td>
<td>4.30 E+05</td>
<td>1.61 E-03</td>
</tr>
<tr>
<td>Zr-93</td>
<td>2.95 E-01</td>
<td>3.0 E-02</td>
<td>8.90 E+04</td>
<td>3.31 E-07</td>
</tr>
<tr>
<td>Nb-93m</td>
<td>1.93 E-01</td>
<td>1.9 E-02</td>
<td>4.30 E+05</td>
<td>4.49 E-08</td>
</tr>
<tr>
<td>Tc-99</td>
<td>2.19 E+00</td>
<td>2.2 E-01</td>
<td>3.80 E+06</td>
<td>5.76 E-08</td>
</tr>
<tr>
<td>Ru-106</td>
<td>2.56 E-02</td>
<td>2.6 E-03</td>
<td>6.50 E+03</td>
<td>3.94 E-07</td>
</tr>
<tr>
<td>Rh-106</td>
<td>2.56 E-02</td>
<td>2.6 E-03</td>
<td>4.30 E+05</td>
<td>5.95 E-09</td>
</tr>
<tr>
<td>Pd-107</td>
<td>1.56 E-02</td>
<td>1.6 E-03</td>
<td>4.30 E+05</td>
<td>3.63 E-09</td>
</tr>
<tr>
<td>Ag-110</td>
<td>7.17 E-10</td>
<td>7.2 E-11</td>
<td>4.30 E+05</td>
<td>1.67 E-16</td>
</tr>
<tr>
<td>Ag-110m</td>
<td>5.39 E-08</td>
<td>5.4 E-09</td>
<td>5.30 E+05</td>
<td>1.02 E-14</td>
</tr>
<tr>
<td>Cd-113m</td>
<td>2.78 E+00</td>
<td>2.8 E-01</td>
<td>4.30 E+05</td>
<td>6.47 E-07</td>
</tr>
<tr>
<td>In-113m</td>
<td>1.36 E-19</td>
<td>1.4 E-20</td>
<td>4.30 E+05</td>
<td>3.16 E-26</td>
</tr>
<tr>
<td>Sn-113</td>
<td>1.36 E-19</td>
<td>1.4 E-20</td>
<td>3.20 E+06</td>
<td>4.25 E-27</td>
</tr>
<tr>
<td>Sn-119m</td>
<td>6.14 E-08</td>
<td>6.1 E-09</td>
<td>4.30 E+05</td>
<td>1.43 E-14</td>
</tr>
<tr>
<td>Sn-121m</td>
<td>6.27 E-02</td>
<td>6.3 E-03</td>
<td>4.30 E+05</td>
<td>1.46 E-08</td>
</tr>
<tr>
<td>Sn-123</td>
<td>1.72 E-16</td>
<td>1.7 E-17</td>
<td>4.30 E+05</td>
<td>4.00 E-23</td>
</tr>
<tr>
<td>Sn-126</td>
<td>1.29 E-01</td>
<td>1.3 E-02</td>
<td>3.30 E+05</td>
<td>3.91 E-08</td>
</tr>
<tr>
<td>Sb-125</td>
<td>0.00 E+00</td>
<td>0.0 E+00</td>
<td>4.30 E+05</td>
<td>0.00 E+00</td>
</tr>
<tr>
<td>Sb-126</td>
<td>1.81 E-02</td>
<td>1.8 E-03</td>
<td>2.50 E+06</td>
<td>7.24 E-10</td>
</tr>
<tr>
<td>Sb-126m</td>
<td>1.29 E-01</td>
<td>1.3 E-02</td>
<td>4.30 E+05</td>
<td>3.00 E-08</td>
</tr>
</tbody>
</table>
Table A3-2. Radionuclide Inventory for the Canister Storage Building Material at Risk Compared with the Category 2 Threshold Quantities. (3 sheets)

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Fuel activity (Ci/MTU)</th>
<th>Material at risk* (100 kg or 0.1 MTU)</th>
<th>Category 2 threshold quantitiesa</th>
<th>Ratio of material to category 2 threshold quantity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Te-123m</td>
<td>1.50 E-21</td>
<td>1.5 E-22</td>
<td>4.30 E+05</td>
<td>3.49 E-28</td>
</tr>
<tr>
<td>Te-125m</td>
<td>0.00 E+00</td>
<td>0.0 E+00</td>
<td>4.30 E+05</td>
<td>0.00 E+00</td>
</tr>
<tr>
<td>Te-127</td>
<td>2.12 E-19</td>
<td>2.1 E-20</td>
<td>4.30 E+05</td>
<td>4.93 E-26</td>
</tr>
<tr>
<td>Te-127m</td>
<td>2.16 E-19</td>
<td>2.2 E-20</td>
<td>1.50 E+05</td>
<td>1.44 E-25</td>
</tr>
<tr>
<td>I-129</td>
<td>5.16 E-03</td>
<td>5.2 E-04</td>
<td>4.30 E+05</td>
<td>1.20 E-09</td>
</tr>
<tr>
<td>Cs-134</td>
<td>6.47 E+00</td>
<td>6.5 E-01</td>
<td>6.00 E+04</td>
<td>1.08 E-05</td>
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<td>Cs-135</td>
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</tr>
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<td>Cs-137</td>
<td>9.66 E+03</td>
<td>9.7 E+02</td>
<td>8.90 E+04</td>
<td>1.09 E-02</td>
</tr>
<tr>
<td>Ba-137m</td>
<td>9.14 E+03</td>
<td>9.1 E+02</td>
<td>4.30 E+05</td>
<td>2.13 E-03</td>
</tr>
<tr>
<td>Ce-144</td>
<td>7.91 E-04</td>
<td>7.9 E-05</td>
<td>8.20 E+04</td>
<td>9.65 E-10</td>
</tr>
<tr>
<td>Pr-144</td>
<td>7.82 E-04</td>
<td>7.8 E-05</td>
<td>4.30 E+05</td>
<td>1.82 E-10</td>
</tr>
<tr>
<td>Pr-144m</td>
<td>9.48 E-06</td>
<td>9.5 E-07</td>
<td>4.30 E+05</td>
<td>2.20 E-12</td>
</tr>
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<td>Pm-147</td>
<td>1.09 E+02</td>
<td>1.1 E+01</td>
<td>8.40 E+05</td>
<td>1.30 E-05</td>
</tr>
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<td>Sm-151</td>
<td>1.02 E+02</td>
<td>1.0 E+01</td>
<td>9.90 E+05</td>
<td>1.03 E-05</td>
</tr>
<tr>
<td>Eu-152</td>
<td>8.45 E-01</td>
<td>8.5 E-02</td>
<td>1.30 E+05</td>
<td>6.50 E-07</td>
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<td>Eu-154</td>
<td>1.13 E+02</td>
<td>1.1 E+01</td>
<td>1.10 E+05</td>
<td>1.03 E-04</td>
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<tr>
<td>Eu-155</td>
<td>1.06 E+01</td>
<td>1.1 E+00</td>
<td>7.30 E+05</td>
<td>1.45 E-06</td>
</tr>
<tr>
<td>Gd-153</td>
<td>5.19 E-19</td>
<td>5.2 E-20</td>
<td>1.40 E+06</td>
<td>3.71 E-26</td>
</tr>
</tbody>
</table>

Actinides

| U-234        | 3.84 E-01             | 3.8 E-02                             | 2.20 E+02                      | 1.75 E-04                                        |
| U-235        | 1.27 E-02             | 1.3 E-03                             | 2.40 E+02                      | 5.29 E-06                                        |
| U-236        | 7.16 E-02             | 7.2 E-03                             | 5.50 E+01                      | 1.30 E-04                                        |
| U-238        | 3.31 E-01             | 3.3 E-02                             | 2.40 E+02                      | 1.38 E-04                                        |
| Np-237       | 4.66 E-02             | 4.7 E-03                             | 5.80 E+01                      | 8.03 E-05                                        |
| Pu-238       | 1.33 E+02             | 1.3 E+01                             | 6.20 E+01                      | 2.15 E-01                                        |
| Pu-239       | 1.73 E+02             | 1.7 E+01                             | 5.60 E+01                      | 3.09 E-01                                        |
| Pu-240       | 1.37 E+02             | 1.4 E+01                             | 5.50 E+01                      | 2.49 E-01                                        |
| Pu-241       | 6.82 E+03             | 6.8 E+02                             | 2.90 E+03                      | 2.35 E-01                                        |
| Pu-242       | 8.71 E-02             | 8.7 E-03                             | 5.50 E+01                      | 1.58 E-04                                        |
Table A3-2. Radionuclide Inventory for the Canister Storage Building Material at Risk Compared with the Category 2 Threshold Quantities. (3 sheets)

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Fuel activity (Ci/MTU)</th>
<th>Material at risk(^a) (100 kg or 0.1 MTU)</th>
<th>Category 2 threshold quantities(^b)</th>
<th>Ratio of material to category 2 threshold quantity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Am-241</td>
<td>4.34 E+02</td>
<td>4.3 E+01</td>
<td>5.50 E+01</td>
<td>7.89 E-01</td>
</tr>
<tr>
<td>Am-242</td>
<td>3.71 E-01</td>
<td>3.7 E-02</td>
<td>4.30 E+05</td>
<td>8.63 E-08</td>
</tr>
<tr>
<td>Am-242m</td>
<td>3.72 E-01</td>
<td>3.7 E-02</td>
<td>5.60 E+01</td>
<td>6.64 E-04</td>
</tr>
<tr>
<td>Am-243</td>
<td>2.78 E-01</td>
<td>2.8 E-02</td>
<td>5.50 E+01</td>
<td>5.05 E-04</td>
</tr>
<tr>
<td>Cm-242</td>
<td>3.08 E-01</td>
<td>3.1 E-02</td>
<td>1.70 E+03</td>
<td>1.81 E-05</td>
</tr>
<tr>
<td>Cm-244</td>
<td>4.47 E+00</td>
<td>4.5 E-01</td>
<td>5.50 E+01</td>
<td>8.13 E-03</td>
</tr>
</tbody>
</table>

\(^a\) Sum of category 2 ratios (inventory material/threshold quantities): 1.85

Note: Radionuclide values are from HNF-SD-SNF-TI-015, 1998, Spent Nuclear Fuel Project Technical Databook, Rev. 6, Fluor Daniel Hanford, Incorporated, Richland, Washington.

*The material at risk used to calculate the final Canister Storage Building hazard category bounds the beyond design basis accident in Section A3.4.3.1.

Category threshold values are from Table A-1 of DOE-STD-1027-92, Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports, U.S. Department of Energy, Washington, D.C.

MTU = metric ton of uranium.
Using 100 kg (0.1 metric ton) UO₂ dispersible particulate as the MAR, the ratios of the constituents of the MAR to the category 2 threshold quantities were calculated and summed to compare with the summation of radionuclide ratios for category 2 threshold criteria. The sum of the ratios of material inventory to category 2 quantity thresholds is 1.85. Since the sum of ratios is greater than one when compared with category 2 threshold criteria, the CSB remains a category 2 facility as identified during the preliminary hazard classification.

A3.3.2.3 Hazard Evaluation. The final CSB facility hazard analysis identified hazards associated with MCO handling and storage operations to be used in the CSB. Standard industrial hazards were identified and removed from the list of facility hazards used to identify DBAs. The results of the hazard analysis identified hazard scenarios for each major area in the CSB. These were used to define the CSB DBAs selected for further analysis in Section A3.4.2.

The external hazards from human-generated threats to CSB operation identified in Section A1.6 involve only those from aircraft activity. Section A1.6 identifies nine active airports within a 24-mile radius of the CSB and defines a maximum credible evaluation basis aircraft impact for the CSB operating deck and the air intake structures. Based on a structural evaluation of this energy impact documented in Assessment of Aircraft Impact on the Canister Storage Building and Cold Vacuum Drying Facility (Beary 1997), the thickness of the operating deck floor and the air intake structure vertical walls are more than adequate to withstand the impact of the evaluation basis aircraft missile. Therefore, the CSB facility is adequately protected from this threat.

Evaluation of the hazards of nearby external human-generated activities that may represent a threat to the facility is an NRC equivalency requirement specified in HNF-SD-SNF-DB-003. Section A1.7 identifies and addresses the threats to the CSB from nearby external facilities.

HNF-SD-SNF-DB-003 does not require that threats from natural phenomena, fire hazards, and nearby facilities be evaluated in Chapter A3.0. Those threats are evaluated elsewhere in the FSAR and summarized here for completeness. As described in detail in Chapter A2.0, the design criteria for natural phenomena were defined based on applicable DOE orders and NRC equivalency requirements as documented in WHC-SD-SNF-DB-009. These design criteria were incorporated into the design and engineering of SSCs as described in Chapter A4.0 (Petersen 1998a). Therefore, credible natural phenomena hazards are addressed in the design of the facility. The CSB fire hazard analysis (HNF-SD-SNF-FHA-002) identified the fire hazards, fire loading criteria, and appropriate requirements for addressing fire hazards during CSB operation based on applicable DOE orders, regulations, and NRC equivalency requirements. The fire hazard analysis evaluated a diesel fire, and a fire from ordinary combustibles that required equipment and controls, fire protection features (i.e., fire detection equipment and sprinklers), and controls on combustibles. The findings of the CSB fire hazard analysis were incorporated into the design and engineering of CSB SSCs and controls (such as fire loads, inspections, and watches). Therefore, credible fire hazards are addressed in the design of the facility. Threats from nearby facilities are identified in Section A1.7 based on applicable DOE orders, regulations, and NRC...
equivalency requirements. Credible threats from nearby facilities are addressed as described in section A1.7.

A3.3.2.3.1 Planned Design and Operational Safety Improvements. This section discusses commitments for the planned major design, MCO handling, or MCO storage improvements for the facility not yet implemented. Because the facility design and accident analysis have been developing in parallel, feedback to the design process has been available. As the hazard evaluation and specific accident analyses progressed, cost-effective modifications that improve safety have been incorporated into the design. For the unrestrained seismic event, the MHM trolley seismic uplift restraints currently exceed the allowable limits in ASME-NOG-1, *Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder).* The design of the MHM will be strengthened to meet ASME-NOG-1 requirements, and this modification will be incorporated before CSB FSAR implementation.

A3.3.2.3.2 Defense in Depth. A summary of fundamental points relevant to the concept of defense in depth is provided in Section 3.3.2.3.2 of the SNF Project FSAR.

Features Chosen to Provide Defense in Depth for the Canister Storage Building.
Defense-in-depth features for the CSB were selected based on a relative ranking of the hazards from the hazard identification process, followed by selection of the safety-class and safety-significant features and TSRs for the DBAs, which are described in Section A3.4.2. Preventive and mitigative features identified in the hazard analysis (HNF-SD-SNF-HIE-001) but not identified in the accident analysis as safety class, safety significant, or TSRs are identified as additional defense-in-depth features. The defense-in-depth features are presented in the tables that accompany each DBA in Section A3.4.2. Administrative features identified in these tables are in addition to those already identified in Chapters A4.0, A5.0, and the programmatic chapters of the SNF Project FSAR (e.g., Chapters 7.0, 8.0, and 11.0).

The first layer of defense in depth at the CSB is facility design. All SSCs are designed in accordance with applicable codes and standards with a high degree of reliability and simplicity. The design encompasses human factors considerations to ensure that operations can be conducted safely. Defense-in-depth features for preventing and mitigating hazards and accidents associated with recovery-related activities and SSCs also have been identified. The overpack storage tube assemblies and the tube vent and purge cart, described in Chapter A2.0, Section A2.5.3, are used in handling MCOs involved in accidents. The overpack storage tubes and the tube vent and purge cart are, therefore, defense-in-depth items for recovery purposes. Abnormal MCOs or MCOs associated with accidents are discussed in Sections A3.3.3 and A3.3.4 and include MCOs received from the CVDF that are out of receipt specifications and those that are dropped, impacted, or sheared at the CSB. The MCOs that are associated with abnormal events will be returned to normal operations as described in Section A3.3.3.

The MCOs associated with accidents will be handled within recovery operations under operations procedures, with the preferred approach being to move the MCO to an overpack storage tube for short-term observation and storage. Use of the overpack storage tube is an
option for recovery. Recovery will be based on analysis and the development of a recovery plan. The containment tent, described in Chapter A2.0, also is involved with potential recovery-related actions.

Safety-Significant Structures, Systems, and Components. Safety-significant SSCs are predominantly required to prevent or mitigate consequences of postulated accident events to the collocated onsite worker. DOE Letter 97-SFD-172 (Sellers 1997) provides a correlation of evaluation guidelines to identification of safety-significant SSCs. In addition, DOE-STD-3009-94 suggests that SSCs be designated as safety significant if they play a key role in defense in depth (or worker safety). The severity of the event being prevented or mitigated and the number of barriers present are provided in DOE-STD-3009-94 as guidance for the identification of defense-in-depth safety-significant SSCs.

Safety-significant defense-in-depth features are described in Chapter A4.0 and include the following:

- MHM hoist and grapple — Provide safe handling of the MCO
- MHM interlocks P2, P6, and P21 — Prevent damage to MCO or operating deck by structural failure of the MHM
- Sampling/weld station shield halves and impact absorber — Protect MCO from drop by crane into sampling/weld station
- Cask lifting yoke — Limit MCO lifting height
- Standard storage tube bottom and intermediate impact absorbers — Mitigate drop impact to MCO for drop into standard storage tube
- Cask receiving impact absorber — Mitigate drop impact to MCO for drop into the cask receiving pit
- MCO centering guide — Limit eccentricity of MCO drop
- Shield hatch and MCO guide assembly — Mitigate drop impact to MCO from drop at cask receiving pit
- Interface guide ring funnel — Mitigate drop impact to MCO from drop in the storage tube
- Standard storage tube lower flange — Mitigate drop impact to MCO from drop into the standard storage tube
MHM seismic restraints (trolley, turret, and bridge) — Prevent damage to operating
deck from MHM, prevent shear of MCO by MHM

MHM rail and rail frogs — Prevent damage to operating deck from MHM, prevent
shear of MCO by MHM.

**Technical Safety Requirements.** TSRs were identified for postulated accident events that
could challenge accident consequence release limits and evaluation guidelines for the offsite public
and collocated onsite worker. TSRs are identified in the individual DBA sections and further
explained in Chapter A5.0. Only when items are elevated to safety significant to provide
significant defense in depth or significant facility worker safety do they warrant TSR coverage.

**A3.3.2.3.3 Worker Safety.** Worker safety for the CSB is ensured by a combination of
design features that reduce exposure to radioactive (e.g., shielding), toxic (e.g., confinement), and
industrial hazards (e.g., auditory or visual warnings), and by institutional practices that, in total,
provide protection of workers from these hazards. Protection of the facility worker from standard
industrial hazards identified for the CSB is achieved through adherence to the institutional safety
programs described in Chapters 7.0, 8.0, 9.0, 11.0, 15.0, and 17.0 of the SNF Project FSAR and
documented in lower-tier documents, such as health and safety plans and job hazards analyses.

Such industrial hazards do not require specific safety-significant SSCs or TSR-level administrative
features. Therefore, in accordance with the guidance of DOE-STD-3009-94, the remainder of
this section deals with protecting workers from those hazards of facility operation that are
exclusive of standard industrial hazards.

The final CSB hazard analysis provides an overview of the major features protecting facility
workers at the CSB (see HNF-SD-SNF-HIE-001). Worker safety features are an integral part of
facility design and operation. The major features of worker protection are identified in
Table A3-3 and are categorized by hazard. The features presented in Table A3-3 are in addition
to those identified as safety-class or safety-significant features in the DBA sections. The features
in the DBA sections protect the facility worker in the same manner in which they protect the
onsite and offsite receptor. Only those hazards not addressed by DBA evaluations are examined
in this section. The hazard energy source or material and hazardous condition are identified in
Table A3-3 along with protective worker safety features, including passive, active, and
administrative features. SSCs or TSRs are identified as safety significant for the CSB based on
their prevention or mitigation of serious impacts to worker safety. These safety-significant
worker safety features are associated with radiation protection and shielding for facility workers
for activities close to the MCO or the cask. The operating deck also plays a significant shielding
role for personnel working in the operating area and is identified as a safety-class feature to
protect the assumptions in the criticality analysis.
### Table A3-3. Hazards and Safety Features Related to Worker Safety. (2 sheets)

<table>
<thead>
<tr>
<th>Worker safety features</th>
<th>Location/checklist entry</th>
<th>Hazard energy source/material</th>
<th>Hazardous condition</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gas cylinders as missiles (determined to present an extremely low risk due to the presence of passive design features that are inherent to the gas cylinders of concern)</td>
<td>TV-F-06</td>
<td>Linear kinetic - pressure vessel blowdown</td>
<td>Gas cylinders are damaged and become missiles, striking personnel</td>
</tr>
<tr>
<td>Gas cylinder design precludes missile hazard (see Chapters A2.0 and A5.0 for design feature details)</td>
<td>SA-F-06</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Gas cylinders are handled in accordance with approved procedures</td>
<td>OA-F-06</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>WS-F-06</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Hydrogen mixture deflagration (determined to present a low risk due to the extremely low likelihood of occurrence)</td>
<td>TV-J-06</td>
<td>Explosives or pyrophorics - hydrogen</td>
<td>MCO damage within cask results in a flammable hydrogen mixture created by hydrogen</td>
</tr>
<tr>
<td>The cask is designed with materials to prevent release from an MCO leak</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>The MCO is designed to prevent release, and is tested for confinement integrity prior to shipment to the CSB</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>The MCO is inerted and dry (minimal free water)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>The operating area is an open space and the HVAC system provides air exchange</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Smoking not allowed within the facility</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Personnel exposure to radioactive material (determined to present a low risk due to extremely low likelihood of occurrence)</td>
<td>SA-H-06</td>
<td>Pressure, volume - pressure vessels (MCO and MCO-transportation cask system)</td>
<td>MCO damage within cask results in the pressurized release of gases within the MCO during cask receipt activities</td>
</tr>
<tr>
<td>Personnel are trained to facility-specific procedures regarding transportation cask shipping, movement, and handling</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>The cask is designed with materials to prevent release from a damaged MCO</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Cask pressure is read before cask lid removal to alert personnel of any suspect MCO condition</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>A load in/load out area enclosure can be placed over the service station pit when higher than normal pressures are detected in the transportation cask. The load in/load out area enclosure, if in place, provides HEPA-filtered confinement in the event of an MCO pressure release (to workers outside the tent).</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Radiation protection program addresses ALARA principles associated with handling activities (see Chapter A7.0 for radiological protection controls)</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
### Table A3-3. Hazards and Safety Features Related to Worker Safety. (2 sheets)

<table>
<thead>
<tr>
<th>Worker safety features</th>
<th>Location/checklist entry</th>
<th>Hazard energy source/material</th>
<th>Hazardous condition</th>
</tr>
</thead>
<tbody>
<tr>
<td>Direct radiation exposure to personnel (determined to present a moderate risk to personnel due to high radiation source and the potential for streaming)</td>
<td>TV-N-01</td>
<td>Ionizing radiation sources (contents of MCO)</td>
<td>Ionizing radiation</td>
</tr>
<tr>
<td>Passive shielding features within the design limit direct radiation exposure to personnel (see Chapters A4.0 and A5.0 for design feature details)</td>
<td>SA-N-01</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>OA-N-01</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>WS-N-01</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>TV-N-03</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Personnel are trained to facility procedures that require shielding installation (as necessary) before associated MCO operations are performed</td>
<td>SA-N-03</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>OA-N-03</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>WS-N-03</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Interlocks and sensors associated with the MHM are present to ensure the skirt is down before hoist operations are performed, and to ensure storage tube plugs are replaced before the MHM moves away from a storage tube position</td>
<td>TV-N-01</td>
<td>Ionizing radiation sources (contents of MCO)</td>
<td>Ionizing radiation</td>
</tr>
<tr>
<td>See Chapter A7.0 for radiological protection controls</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

ALARA = as low as reasonably achievable.
CSB = Canister Storage Building.
HEPA = high-efficiency particulate air (filter).
HVAC = heating, ventilation, and air conditioning.
MCO = multi-canister overpack.
MHM = multi-canister overpack handling machine.
Safety-significant worker safety features are described in Chapter A4.0 and include the following:

- Standard storage tube plugs — Provide shielding to reduce radiation dose to worker
- Overpack storage tube plug — Provide shielding to reduce radiation dose to worker
- Shield hatch and MCO guide assembly — Provide shielding to reduce radiation dose to worker
- Sampling/weld station shield halves — Provide shielding to reduce radiation dose to workers
- Sampling/weld station center shield plate — Provide shielding to reduce radiation dose to workers
- MCO shield plug shielding — Provide shielding to reduce radiation dose to workers
- Transportation cask shielding — Provide shielding to reduce radiation dose to workers
- MHM fixed shielding — Provide shielding to reduce radiation dose to workers
- MHM interlock P2 — Protect worker against direct radiation exposure
- Sampling hood exhaust system — Protect worker from injury caused by flammable mixtures
- Sampling piping confinement system — Protect worker from injury caused by flammable mixtures

The TSR controls identified in the analysis of the DBAs in Section A3.4.2, in conjunction with the safety features identified in the hazard analysis and institutional programs, are adequate to ensure worker safety. These worker safety TSR controls include the following:

- Standard and overpack storage tube plug placement for direct radiation shielding
- P2 interlock operability to limit personnel exposures
- Sampling/weld station temporary shielding (shield halves and center shield plate), operator insertion of shielding, or verification that shielding is in place
- Shield hatch and MCO guide assembly shielding placement for direct radiation shielding
A3.3.2.3.4 Environmental Protection. The hazard to the environment from CSB operations involves the potential release of contaminants. The release pathway for these contaminants is only via the air to the boundaries and receptors discussed in Section A1.3.1.3. No liquid release hazards or accidents have been identified and no contaminant releases to the ground or groundwater are involved for the CSB. In addition, where liquid releases (such as water from fire sprinkler system in the support building, battery acid used in carts) are possible, they are contained by the design of the facility.

Based on the CSB design and operating information, no use of toxic chemicals has been identified. The toxicological hazards of the radionuclide inventory have been reviewed. As described in Section 3.4.1.1 of the SNF Project FSAR, the radiological guidelines have been found to be more limiting than the toxicological guidelines for the release of SNF particulate. The SNF particulate is primarily oxides of uranium (with fission products), which are not expected to change under current accident conditions. Potential consequences, including offsite releases, and required prevention and mitigation features are discussed in Section A3.4.2. Implementation of the prevention and mitigation features will prevent large releases that could have significant environmental impact.

The project features that protect the onsite collocated worker and the offsite public against radiological exposure also serve to prevent and mitigate radiological release to the environment. In addition, sitewide programs for environmental monitoring provide for assessment of the impact of facility releases. Normal CSB MCO handling or storage activities are expected to have a minor impact on the local and regional environment, as noted in WHC-SD-SNF-TI-013, K Basins Environmental Impact Statement Technical Input Document.

A3.3.2.3.5 Accident Selection. The methodology for the selection of DBAs is specified in DOE-STD-3009-94. DBAs are to be selected so that the range of accident scenarios analyzed in the accident analysis represents a complete set of representative and bounding conditions. This is a common requirement among the SNF Project facilities and is described in Section 3.3.2.3.5 of the SNF Project FSAR. The list of six candidate accidents resulting from the hazards binning process for the CSB facility is presented in Table A3-4. The table also contains all identified category S3 and S2 events bounded by each candidate accident. All of these events, and the controls selected for their prevention and mitigation, are described in Sections A3.4.2.1 through A3.4.2.6.
Table A3-4. Binned Listing of Candidate Accidents. (2 sheets)

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Risk ranking</th>
<th>Release or change energy</th>
<th>Reference designator</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Mechanical damage of MCO (Section A3.4.2.1)</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Possible mechanical damage of MCO due to a drop</td>
<td>5</td>
<td>Medium</td>
<td>TV-G-13</td>
</tr>
<tr>
<td>Possible mechanical damage of MCO due to a shear</td>
<td>7</td>
<td>Low</td>
<td>SA-E-07</td>
</tr>
<tr>
<td>Possible mechanical damage of MCO due to an impact other than drop or shear of MCO or cask-MCO</td>
<td>5</td>
<td>Medium</td>
<td>SA-G-03a, -03c</td>
</tr>
<tr>
<td><strong>Gaseous release from the MCO (Section A3.4.2.2)</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Pressurized release from MCO</td>
<td>7</td>
<td>Medium</td>
<td>WS-G-04b, -06b, -07b</td>
</tr>
<tr>
<td><strong>MCO internal hydrogen deflagration (Section A3.4.2.3)</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Hydrogen deflagration</td>
<td>8</td>
<td>High</td>
<td>SA-J-06a</td>
</tr>
</tbody>
</table>

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March 2000
Table A3-4. Binned Listing of Candidate Accidents. (2 sheets)

<table>
<thead>
<tr>
<th>Candidate accident</th>
<th>Risk ranking&lt;sup&gt;a&lt;/sup&gt;</th>
<th>Release or change energy&lt;sup&gt;b&lt;/sup&gt;</th>
<th>Reference designator</th>
</tr>
</thead>
<tbody>
<tr>
<td>MCO external hydrogen deflagration (Section A3.4.2.4)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>External deflagration</td>
<td>7</td>
<td>High</td>
<td>WS-L-11</td>
</tr>
<tr>
<td>Thermal runaway fuel reactions inside the MCO (Section A3.4.2.5)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel reaction with water</td>
<td>8</td>
<td>Medium</td>
<td>OU-R-01</td>
</tr>
<tr>
<td></td>
<td>6</td>
<td>High</td>
<td>SA-J-10b</td>
</tr>
<tr>
<td></td>
<td>6</td>
<td>High</td>
<td>OA-J-10b</td>
</tr>
<tr>
<td></td>
<td>6</td>
<td>High</td>
<td>WS-J-10b</td>
</tr>
<tr>
<td>Violations of design temperature criteria (Section A3.4.2.6)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Violation of design temperature criteria</td>
<td>6</td>
<td>Medium</td>
<td>VL-B-07, -10, -11</td>
</tr>
</tbody>
</table>

<sup>a</sup>The risk ranking is derived from methodology found in DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, which correlates the consequence–frequency pairs assigned by the hazard analysis (HNF-SD-SNF-HIE-001) to a single-scale risk ranking using a figure reproduced in Figure 3-1 of the SNF Project FSAR. Events identified by a dash (-) were considered SI events in the hazard analysis but were initially considered of particular concern to the SNF Project and therefore were to be evaluated in more detail.

<sup>b</sup>Definition and use of energy release categories (high, medium, low) are based on guidance and examples in DOE-STD-3009-94.

Energy was considered that could damage an MCO — falling onto the deck was viewed as higher energy than falling into the cask receiving or sampling/weld pit with impact absorbers present; falling into the tube with impact absorbers present was viewed as higher energy than falling into the cask receiving or sampling/weld pit with an impact absorber present.

Before detailed analysis was performed, the hazard evaluation identified WS-H-06b as a serious hazard to be evaluated. Subsequent detailed analysis has shown that thermal runaway reactions are not possible at the CSB given limitation of water and resulting temperature.

MCO = multi-canister overpack.
It should be noted that early in the SNF Project history, DOE and the SNF Project had decided that the functions of vault structural capability and vault cooling were important safety-class functions. These defined safety functions of the vault led to the classification of the vault and related supporting structures. The safety-class seismic response spectra, as defined in WHC-SD-SNF-DB-004, *Spent Nuclear Fuel Project Seismic Design Criteria, NRC Equivalency Evaluation Report*, were used for the design of the CSB structures.

From the relevant sections of Letter 97-SFD-227, *Contract No DE-AC06-96RL13200 — Conditional Acceptance of Phase Three, Canister Storage Building Safety Analysis Report* (SAR), WHC-SD-SNF-RPT-004, *Systems* (Hansen 1997), the bases for DOE, Richland Operations Office, authorization of construction of the CSB were established. This included identification of seismic criteria. DOE's review, as documented in Letter 97-SFD-227, focused on the technical accuracy and completeness of HNF-SD-SNF-RPT-004, Revision 7B, and supporting references. It was a specific intent of the DOE review to ensure that the design and construction of the structure and systems would meet applicable DOE criteria and NRC requirements.

To meet the aggressive completion schedule for the SNF Project, DOE decided to implement a “staged safety analysis report approach” that waived the requirement for a preliminary safety analysis report prior to construction. Instead, a single, staged FSAR was to be produced. To implement the staged safety analysis report approach, DOE determined the basis for authorizing the first phase of construction. As of the date of Letter 97-SFD-227 (Hansen 1997), DOE had performed a number of reviews that provided the basis for the authorization of construction. This authorization included the at-grade and below-grade portions of the CSB, including the load-in/load-out area, intakes, stack foundation, service building foundation, the superstructure, and the CSB systems, with the exception of the MHM.

Further, Letter 97-SFD-227 (Hansen 1997) identified the site characteristics necessary for understanding the SNF storage facility environs important to the safety basis of the facility. The letter found the sections in the safety analysis report related to hydrology, geology, and natural phenomena threats were acceptable.

Based on this phased approach to regulatory approvals, design bases, design, and construction, it is clear that the seismic design criteria approach was the basis for approval and construction. The details of these seismic design bases are described in Section 4.4 of Letter 97-SFD-227 (Hansen 1997). A technical basis was established for the geology and natural phenomena hazards. These hazards were used as the basis for seismic and natural phenomena criteria for the design and construction of the CSB structures and systems, except the MHM.

In regard to hazard and accident analyses, the methodology used and the analyses performed and documented in HNF-SD-SNF-RPT-004, Revision 7B, are consistent with the analysis reviewed in Letter 97-SFD-227 (Hansen 1997). In particular, the letter noted that “the results from the previous PHA [preliminary hazard analysis] were grouped into six accident types and a representative accident for each group was determined to be bounding. This is still the case
for the new HA, but the types of accident phenomena have changed.” None of the identified representative accidents is a seismic DBA. The letter concluded that “based on its judgement, the review team feels the proposed accident methodology is adequate for this phase of construction.”

As such, Letter 97-SFD-227 did not require the application of hazard and accident analysis methodology for the determination of the seismic performance of SSCs. Rather, Letter 97-SFD-227 accepted that these were defined by seismic design criteria set during the early phased FSAR approach.

A3.3.3 Abnormal Events for the Canister Storage Building Facility

Abnormal events are operating conditions resulting from situations outside of normal operations, where normal operations are defined by process flow diagram, system design descriptions, and operation and maintenance procedures. These abnormal events encompass malfunctions of systems, operating upset conditions, or operator error. Abnormal events are expected to occur annually or several times during the lifetime of the facility. Abnormal events may impact operational or programmatic schedules; however, the consequences of the events are near zero or are standard industrial hazards that may include worker radiation exposure.

This section addresses the identification and analysis of abnormal events having the potential to occur at the CSB. Using the process described in Chapter 3.0 of the SNF Project FSAR, hazardous conditions and associated accident conditions and events meeting the abnormal event profile were identified from the hazard analysis (HNF-SD-SNF-HIE-001). The hazardous conditions having common failures or impacts on facility operations were grouped into bins. Appendix A3A.0 presents the abnormal event bins, and lists events in each bin. Two bins, having distinct initiators and operational consequences, were identified at the CSB: (1) MCO 'suspect' condition and (2) loss of ability to normally handle or store cask-MCO. Bin 2 has the following four parts: 2A, electrical/mechanical failure; 2B, structural; 2C, fire; and 2D, errors. For each of these event bins, the descriptive information includes the event, the cause of the event, the consequence of the event, the means of detection for the event, and corrective actions.

The following sections provide a description of the CSB abnormal event analysis.

A3.3.3.1 Multi-Canister Overpack "Suspect" Condition. Events in bin 1 have an impact on CSB operations because of the uncertainty of the MCO structure or integrity, the internal geometry of the SNF within the MCO, or the chemical or thermal status of the SNF in the MCO. During the normal course of operations, if an event has occurred that raises uncertainty about the MCO condition, then an abnormal event has occurred and will be addressed as described below. However, it should be noted that the postulated causes of events in this bin can be the same as the postulated causes of some CSB DBAs. Consequences from these events, therefore, are prevented and/or mitigated as required by DOE orders. These events were included in the abnormal event analysis because they could occur during normal operations and not result in, or propagate, accident conditions.
Postulated Causes of the Event. Events in bin 1 involve:

- A higher than expected pressure or temperature
- An impact to the MCO during handling within the CSB
- Mishandling or misloading the MCO.

Receipt of an out-of-specification MCO could result from improper processing or shipping preparations at the K Basins or CVDF, and/or chemical reactions within the MCO occurring at a rate greater than anticipated. An MCO having a high pressure on receipt raises uncertainty as to proper handling, sampling, and storage procedures at the CSB.

Both an impact to the MCO during handling and mishandling an MCO within the CSB raise uncertainties regarding the MCO's structure and integrity, or the internal geometry of the SNF within the MCO. These events lead to uncertainty as to proper handling, sampling, and storage procedures at the CSB.

Means of Detection. Events in bin 1 are detected by indications or measurements of out-of-specification pressures in the cask-MCO, and direct operator observation of an impact to the MCO or the mishandling of an MCO.

Consequences. Events in bin 1 could impact the facility or project schedules and could result in an increase of facility occupational exposure totals. Receiving an out-of-specification MCO would result in facility delays caused by investigations to determine the reasons for the out-of-specification situation and special handling requirements for handling the cask or the MCO. Similarly, impacts, mishandling, and misloading of the MCO would require an investigation to determine the MCO condition and special handling requirements, if any, for continued MCO handling and storage. Special handling requirements could result in increased radiological exposure totals for the facility. Individual workers would not exceed radiological protection and ALARA (as low as reasonably achievable) program limits.

Corrective Actions and Return to Normal Operations. Immediate operational responses to bring events in bin 1 to safe and stable states are described in standard operating procedures, alarm response procedures, or emergency response procedures. Operating personnel are routinely trained in the use of these procedures. Once the event has been stabilized, further diagnosis will be required. Facility management, or a recovery team, will lead the investigation and develop a recovery plan following procedures written for anticipated abnormal events. Procedures addressing anticipated abnormal events specify MCO handling requirements and actions to return the facility to normal operations. Specific equipment operations during recovery are expected to be the same as those contained in the standard operating procedures. If necessary, the recovery team can specify additional MCO handling requirements normally not contained in the standard operating procedures. The important actions to be addressed by procedures and the recovery team are shown in Appendix A3A.0, Table A3A-1.
A3.3.3.2 Loss of Ability to Normally Handle or Store Cask–Multi-Canister Overpack.

Events in bin 2 impact CSB operations because of the inability of the MCO or the cask–MCO handling equipment to perform their handling functions. Bin 2 has four parts: electrical/mechanical failure; structural; fire; and errors. Until the equipment or structure is repaired or returned to operating status, MCO handling is halted within the CSB.

Postulated Causes of the Event. Events in bin 2 involve the functional loss of various MCO handling equipment within the CSB. Causes can include the following:

- Loss of power
- Mechanical failures in the MHM, receiving crane, or other handling systems
- Failure of vehicles associated with MCO handling
- Fires in the CSB
- Severe natural phenomena affecting MCO transport to the CSB or the CSB operating environment that will affect handling operations (e.g., extremely high or low temperatures in the operating area)
- Leakage of water into the facility that will violate the CSB requirements for the control of criticality
- Operator errors that would impact the functionality of the MCO handling equipment.

Means of Detection. Events in bin 2 are detected by instrumentation indications and direct operator observation of a failure of handling equipment to function, visual indications and audible soundings from alarms (including fire alarms or the sound of failed equipment), or notification by CSB maintenance personnel or other Hanford Site or offsite entities.

Consequences. Events in bin 2 primarily result in (1) schedule delays for placing the MCO into storage, (2) the potential for increased occupational radiological exposures to workers, and (3) the potential for programmatic delays to the SNF Project. Failures in handling equipment can lead to facility delays caused by investigations to resolve the reasons for the equipment failures and to determine the special requirements for handling the cask or the MCO. Special handling may be required, which could increase radiological exposure. In the event of a fire, equipment and personnel could be impacted by smoke, heat, and chemical vapors.

Corrective Actions and Return to Normal Operations. Immediate operational responses to bring events in bin 2 to safe and stable states are described in standard operating procedures, alarm response procedures, or emergency response procedures. Operating personnel are routinely trained in the use of these procedures. Once the event has been stabilized, further diagnosis could be required. Facility management, or a recovery team, will develop a recovery
plan following procedures written for anticipated abnormal events. Procedures addressing anticipated abnormal events specify MCO handling requirements and actions to return the facility to normal operations. The failed or damaged equipment would be repaired and tested before return to operation. For several of these events, the facility would need to be cleaned before resumption of normal operations. Specific equipment operations during recovery are expected to be the same as those contained in the standard operating procedures. If necessary, the recovery team can specify additional MCO handling requirements not normally described in the standard operating procedures. The important actions to be addressed by procedures and the recovery team are shown in Appendix A3A.0, Table A3A-1.

A3.4 ACCIDENT ANALYSIS

This section presents the methodology used to develop the DBAs identified in Section A3.3 and the quantified consequences of those events. It also presents the safety-class and safety-significant SSCs and TSRs necessary to protect the offsite public and onsite workers. For each DBA, the following standard topics are discussed:

- Scenario development
- Source term analysis
- Consequence analysis
- Comparison to guidelines
- Summary of safety SSCs and TSRs.

The hazard and accident analyses focused on addressing operational radiological related hazards and design basis events.

- Mechanical damage of MCO (Section A3.4.2.1)
- Gaseous release from the MCO (Section A3.4.2.2)
- MCO internal hydrogen deflagration (Section A3.4.2.3)
- MCO external hydrogen deflagration (Section A3.4.2.4)
- Thermal runaway reactions inside the MCO (Section A3.4.2.5)
- Violation of design temperature criteria (Section A3.4.2.6)

The consequences associated with each of these bounding DBAs are summarized in Table A3-5.

A3.4.1 Methodology

This section identifies methods, assumptions, or methodology used to quantify the consequences of the DBAs. Methods, assumptions, or methodology used to quantify the consequences of DBAs that are common or generic to all the SNF Project facilities at the CVDF, or CSB are described in Chapter 3.0 of the SNF Project FSAR.
Table A3-5. Summary of Consequences for Bounding Design Basis Accidents.

<table>
<thead>
<tr>
<th>Accident category</th>
<th>Section</th>
<th>Offsite consequences</th>
<th>Onsite consequences</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Release limit (rem)</td>
<td>Unmitigated (rem)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>and frequency</td>
<td>and frequency</td>
</tr>
<tr>
<td>Mechanical damage of MCO</td>
<td>A3.4.2.1</td>
<td>5.0 (unlikely)</td>
<td>2.2 E-03</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>10 (unlikely)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Gaseous release from MCO</td>
<td>A3.4.2.2</td>
<td>0.5 (anticipated)</td>
<td>5.1 E-05</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>1.0 (anticipated)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>4.5 E-02</td>
</tr>
<tr>
<td>MCO internal hydrogen deflagration</td>
<td>A3.4.2.3</td>
<td>5.0 (unlikely)</td>
<td>3.4 E-03</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>10 (unlikely)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>3.0</td>
</tr>
<tr>
<td>MCO external hydrogen deflagration</td>
<td>A3.4.2.4</td>
<td>0.5 (anticipated)</td>
<td>4.3 E-04</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>1.0 (unlikely)</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>0.38</td>
</tr>
<tr>
<td>MCO thermal runaway reaction</td>
<td>A3.4.2.5</td>
<td>Beyond extremely unlikely</td>
<td>Beyond extremely unlikely</td>
</tr>
<tr>
<td>Violation of design temperature criteria</td>
<td>A3.4.2.6</td>
<td>Beyond extremely unlikely</td>
<td>Beyond extremely unlikely</td>
</tr>
</tbody>
</table>


MCO = multi-canister overpack.

A3.4.1.1 Scenario Development. Accident scenarios are developed for the bounding events in each accident category (see Table A3-5).

A3.4.1.2 Source Term. The common SNF Project methodology and assumptions used in developing the source term are described in Chapter 3.0 of the SNF Project FSAR. In addition, the CSB accident analyses have used the following methodology for dealing with the uncertainty associated with the source term.

In calculating consequences in safety analysis, a number of phenomenological uncertainties are usually associated with the calculation factors. While it is not appropriate to ignore these uncertainties and use only best estimates, it also is not appropriate or meaningful to compound all the uncertainties and use an ultraconservative result that has no real practical meaning. The following approach has been developed to acknowledge the individual uncertainties in the calculation factors and to combine them into an overall uncertainty factor that can be applied to the best-estimate results to arrive at a bounding value at a predetermined confidence level.
Uncertainty (or error) factors and best estimates that are conservative but credible can be developed for many of the variables used in accident analyses. Multiplying each best estimate by an error factor yields a bounding value for that factor. If the uncertainties in all these factors are multiplied together, however, the result is unrealistically high because it is exceedingly unlikely that all these factors would simultaneously be at their most pessimistic bounding value. Therefore, a better approach is to calculate a source term using nominal (or median) values of the uncertain parameters and apply an appropriate uncertainty or error factor to the result. The method for combining uncertainty factors to arrive at an overall upper bound value is developed in the following paragraphs.

Assume a Lognormal Distribution for Each Variable. An upper bound value for a variable derived as the product of a number of other parameters can be determined using a simple sum-of-the-products-of-the-factors model. By using lognormal distributions, error factors for each of the individual variables can be combined into an overall error factor to obtain an upper bound value (e.g., 95th percentile).

The usefulness of the lognormal distribution comes from the central limit theorem, which states that the product of \( n \) independent random variables is a lognormally distributed random variable for large \( n \) (Apostolakis 1974). For example, the MAR is treated as the product of a number of processing parameters, such as fuel surface area, reaction rate, and processing time. It is therefore reasonable to describe the MAR as a variable with a lognormal distribution.

The ratio of the upper bound value to the median value (i.e., 50th percentile) is the error factor. The application of this type of bounding error factor is sufficiently conservative for determining the bounding consequences for an accident analysis as well as for comparing with the radiological evaluation guidelines and release limits.

Identify Nominal and Bounding Values and Assign a Percentile to the Bounding Value. For each identified parameter of the product used in the calculation, an estimate of the nominal or median value, \( X_{50\%} \) (i.e., the value with 50% chance of being exceeded) is determined. An estimate of the upper bounding value, say \( X_{95\%} \) (i.e., the value with 5% chance of being exceeded) also is identified. An error factor is calculated by dividing the bounding value by the nominal value of the parameter (i.e., \( EF = \frac{X_{95\%}}{X_{50\%}} \)).

Calculate the Standard Normal Variable for Each Percentile. A confidence level is assigned to the bounding value for each identified parameter. The standard normal variable corresponding to the 95% confidence interval is 1.645. It would be 1.0 for the 84% confidence level, 1.28 for the 90% confidence level, and 2.32 for the 99% confidence level.

Estimate the Overall Median. The overall nominal value is the product of the individual nominal values.
Estimate the Overall Error Factor. The overall error factor for the 95% confidence level is the natural antilogarithm of the product of 1.645 times the square root of the sum of the squares of the natural logarithm of the individual error factors.

Estimate the Ninety-Fifth Percentile Upper Bound. The 95% upper bound value is the product of the overall nominal value times the overall error factor.

The use of this method to estimate the bounding value is a reasonable approximation for estimating uncertainty. This method is 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 uncertainties in these factors are uncorrelated.

In the following example, this method is applied to determining the amount of particulate suspended inside an MCO by the shock impact of an MCO drop. The estimated amount of suspended material depends on the airborne release fraction (ARF), respirable fraction (RF), MAR, and the leak path factor (LPF) for the MCO. Statistical descriptors for each of these quantities are estimated from the literature or from SNF Project documentation as appropriate.

If an MCO were dropped, the impact of the drop would shock and suspend some amount of particulate matter within the MCO. Unless the MCO were also breached, none of this material would be released. However, for a breached MCO, the gas from the blowdown of the MCO would carry some fraction of the suspended material outside the MCO. The amount of respirable particulate released can be estimated using the following equation:

\[ M_{\text{released}} = \text{ARF} \times \text{RF} \times \text{MAR} \times \text{LPF}_{\text{MCO}} \]

where

- \( M_{\text{released}} \) = mass of material released
- ARF = airborne release fraction
- RF = respirable fraction
- MAR = material at risk
- LPF_{MCO} = leak path factor for MCO.

The ARF is the fraction of the particulate powder inside the MCO (UO\(_2\) particulate) that is suspended by the drop. The RF is the fraction of the suspended powder that is respirable. DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions/Rates for Nonreactor Nuclear Facilities* (Section 4.4.3.3, "Impact"), provides a measured median ARF of \( 4 \times 10^{-4} \) and RF of 0.2 from the impact of structural debris on powder. The recommended ARF is \( 1.0 \times 10^{-3} \), used in conjunction with an RF of 1.0, as the bounding value (assumed to be the 95th percentile). The resuspension of material in the MCO, either by drop impact, MCO handling, or internal or external hydrogen deflagrations, is assumed to use these bounding values.
The amount of material available for release depends on the amount of particulate mass generated in the MCO during processing. This quantity is a function of processing times and temperatures as well as of the condition of the fuel in the MCO (e.g., exposed surface area of fuel). HNF-SD-SNF-TI-015, *Spent Nuclear Fuel Project Technical Databook*, provides two estimates of this quantity: one for the safety basis, which is considered the bounding value, and the other for the design basis, which is considered the nominal value. When the MCO is received at the CSB, the safety basis value is 26.3 kg UO$_2$ (23.1 kg uranium) and the design value is 0.613 kg UO$_2$ (0.54 kg uranium) (HNF-SD-SNF-TI-015). After 40 years of storage at the CSB, the safety basis value is 34 kg UO$_2$ (30 kg uranium) and the design basis is 2.1 kg UO$_2$ (1.85 kg uranium) (HNF-SD-SNF-TI-015). The 40-year design basis value is used as the nominal value and the 40-year safety basis value is assumed to be the 99th percentile value.

The pressure in the MCO at the start of the blowdown determines the fraction of the suspended particulate that is released. The initial pressure varies from 1.50 atm (corresponding to the helium fill pressure of 22 lb/in$^2$ absolute at the CVDF) to a maximum pressure of 5.2 atm (as defined in the Technical Databook [HNF-SD-SNF-TI-015]). To allow for some additional margin in the calculation, a pressure of 2.0 atm is considered nominal (50th percentile) and corresponds to an LPF of 0.5, while a pressure of 6.0 atm is considered bounding (95th percentile) and corresponds to an LPF of 0.83.

Table A3-6 shows factors used in calculating the overall bounding value of material suspended, 1.2 g. If the individual bounding values for each of the factors were to be multiplied together, the product of the individual bounding values is 25 g. This is equivalent to an uncertainty or error factor of over 330 (equivalent to a confidence level of 99.97%) compared with the calculated uncertainty factor of 15.7 (for the 95th percentile). This methodology is applied to the calculation of the source term for each of the DBAs as appropriate.

A3.4.1.3 Consequence Analysis. Radiological inhalation dose consequences for each accident analyzed are based on the following factors:

- Mass of respirable airborne material released (M)
- Material at risk (MAR)
- Respirable fraction (RF)
- Airborne release fraction (ARF) or airborne release rate (ARR)
- Leak path factor (LPF)
- Atmospheric transport factor ($\chi/Q'$)
- Breathing rate (BR)
- Dose per unit mass of uranium (UD)
- Duration of exposure.

Some of these parameters are CSB-specific and are discussed below; others are based on common methods, assumptions, or methodology for the SNF Project and are fully described in Chapter 3.0 of the SNF Project FSAR.

<table>
<thead>
<tr>
<th>Factor</th>
<th>Individual median values</th>
<th>Individual bounding values</th>
<th>EF</th>
<th>In(EF)</th>
<th>Ln(EF)/SNV</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Bounding</td>
<td>Percentile</td>
<td>SNV</td>
<td></td>
</tr>
<tr>
<td>MAR</td>
<td>1.85 kg U</td>
<td>30 kg U</td>
<td>99</td>
<td>2.326</td>
<td>16.22</td>
</tr>
<tr>
<td>ARF</td>
<td>4.00 E-04</td>
<td>1.00 E-03</td>
<td>95</td>
<td>1.645</td>
<td>2.50</td>
</tr>
<tr>
<td>RF</td>
<td>0.20</td>
<td>1.00</td>
<td>95</td>
<td>1.645</td>
<td>5.00</td>
</tr>
<tr>
<td>LPF_{MCO}^*</td>
<td>0.50</td>
<td>0.83</td>
<td>95</td>
<td>1.645</td>
<td>1.67</td>
</tr>
</tbody>
</table>

Square root \left\{\text{sum of} \ln(EF)/\text{SNV}\right\} = 1.673

Overall EF(95%) = \exp(1.645 \times \text{square root} \left\{\text{sum of} \ln(EF)/\text{SNV}\right\}^2) = 15.7

Overall median = \text{product of the individual median values} = 7.4 \times 10^{-2} \text{ g}

Overall bounding (95%) = EF(95\%) \times \text{overall median} = 1.2 \text{ g}

Product of individual bounding values = 25 \text{ g}

Percentile of product of individual bounding values = 99.97

Note: The 1.645 is the standard normal variable corresponding to the 95% upper confidence limit. For the 99% upper confidence limit, the corresponding value is 2.326.

\*LPF_{MCO} = (\text{MCO pressure} - 1 \text{ atm}) / (\text{MCO pressure}). At a nominal MCO pressure of 2.0 \text{ atm}, the LPF_{MCO} is 0.5, and at a bounding MCO pressure of 6.0 \text{ atm}, the LPF_{MCO} is 0.83.

ARF = airborne release fraction.
EF = error factor (i.e., bounding value divided by nominal value).
ln(EF) = natural log of EF.
LPF = leak path factor.
MAR = material at risk.
MCO = multi-canister overpack.
RF = respirable fraction.
SNV = standard normal variable.

The mass of respirable airborne material released (M) is determined by the specific CSB accident scenario. The quantity (M) is a function of the MAR and the RF, and the LPF of any passive structural enclosure that may cause deposition of an airborne release before the release enters the atmosphere. The LPF is based on a time-integrated calculation of aerosol deposition within and release from an enclosure of given dimensions with specified leakage area, pressure, and temperature differentials. The specific value of each parameter is determined in the individual DBA analysis and based on the physical phenomena of the accident; thus they are specific to CSB.

The atmospheric transport factor (\chi/Q') is based on specific release conditions (e.g., ground level or elevated, long or short duration) and the receptor's distance from the release.
While the methodology is common to the SNF Project, the atmospheric transport factor is the
time-integrated normalized air concentration at the receptor's location, which is a measured
distance from the CSB. The transport factor includes the dilution of an airborne contaminant
caused by atmospheric mixing and turbulence. The air transport values used in this report have
been generated using GXQ, which is described in WHC-SD-GN-SWD-30002, GXQ Program
User's Guide. GXQ is a FORTRAN program for calculating atmospheric dispersion using site-
specific wind data. It uses the Gaussian straight line model for both instantaneous and continuous
releases. Several models are available that modify parameters within the Gaussian plume model to
account for phenomena such as plume depletion, building wake, plume meander, gravitational
settling, and plume rise. The building wake model from NRC Regulatory Guide 1.145,
Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear
Power Plants, was used to model CSB building effects. The treatment of site wind data is also
subject to user controls to allow various frequencies of exceedance to be computed. GXQ is
intended to be used by individuals who understand the limits and applicability of the models
implemented. According to WHC-SD-GN-SWD-30003, GXQ Program Verification and
Validation, the program has been tested and verified to implement its calculational models
correctly. Table A3-7 contains the atmospheric transport values used to determine onsite and
offsite consequences.

<table>
<thead>
<tr>
<th>Receptor location description</th>
<th>Air transport factorsa</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Acute</td>
</tr>
<tr>
<td></td>
<td>Less than 1 hourb</td>
</tr>
<tr>
<td>Onsite worker (100 m)</td>
<td>3.41 E-02</td>
</tr>
<tr>
<td>Onsite worker (100 m) with building wake effect</td>
<td>1.14 E-02</td>
</tr>
<tr>
<td>Highway 240 (9,280 m W)</td>
<td>2.36 E-05</td>
</tr>
<tr>
<td>Hanford Site boundary (17, 390 m E)</td>
<td>1.30 E-05</td>
</tr>
</tbody>
</table>

a Units for these values are seconds per cubic meter. In all cases the releases are assumed to be point sources at ground level to maximize the dose consequences.
b No adjustment for plume meander (HNF-SD-SNF-TI-059, 1998, A Discussion on the
Methodology for Calculating Radiological and Toxicological Consequences for the Spent Nuclear
Fuel Project at the Hanford Site, Rev. 1, Fluor Daniel Hanford, Incorporated, Richland,
Washington.).

Atmospheric transport factors were calculated using methods found in NRC Regulatory
Guide 1.145. In each wind direction the observed frequencies of particular wind speed and
stability class combinations were used to compute $\chi/Q$. For the accident analysis, the higher of
either the 99.5% sector-dependent or the 95% overall value was used. This was repeated for all
16 compass directions to determine the worst-case location.

Exposures to the collocated worker onsite are calculated for the individual at the 100-m
location. The risk evaluation guidelines apply to this individual. For assessment purposes, DOE
has directed (Sellers 1996) that the Hanford Site boundary be considered the location of the
offsite receptor. Consequences for a receptor located on Highway 240 are included for
information only.

None of the accidents analyzed in this document adjusts the air transport factors for the
elevation of the release above ground level (the stack is not high enough relative to the operation
building to justify using elevated release). It is always conservative to ignore stack effects
because the stack effect serves to disperse the release such that the collocated worker and the
offsite receptor will receive a lower estimated dose if the effect is included. As an additional
conservatism, all accidents were evaluated using the air transport factors calculated for less than
1 hour (HNF-SD-SNF-TI-059). Section 1.4.1.2.8 of the SNF Project FSAR provides additional
information on the calculation of the air transport factors.

The breathing rate (BR) depends on individual activity factors and exposure duration. This
methodology is common to the SNF Project facilities and is described in Chapter 3.0 of the SNF
Project FSAR.

The dose per unit mass of uranium (UD) is the 50-year dose commitment for all relevant
exposure pathways per gram of uranium released (HNF-SD-SNF-TI-059). The major radiation
exposure pathway for the identified accidents is inhalation of radioactive material. This
methodology is common to the SNF Project facilities and is described in Chapter 3.0 of the SNF
Project FSAR.

A3.4.1.4 Comparison to Guidelines. The DOE-recommended radiological release limits and
risk evaluation guidelines (Sellers 1997) are applied across the SNF Project and are described in
Chapter 3.0 of the SNF Project FSAR. The guidelines are consequence (see Section A3.4.1.3)
and frequency based. The determination of frequency is based on methodology common to the
SNF Project facilities and is described in Chapter 3.0 of the SNF Project FSAR. SNF-4042
documents the accident frequencies and bases for determining accident frequencies for the CSB
DBAs.

A3.4.1.5 Safety Structures, Systems, and Components. "Safety class" and "safety
significant," as related to the SSCs, are defined consistently for the SNF Project in Chapter 3.0 of
the SNF Project FSAR.

For each accident scenario, the airborne radiological dose calculated using the methods
described here and in Chapter 3.0 of the SNF Project FSAR is compared with the appropriate
onsite evaluation guidelines and offsite limits from Letter 97-SFD-172 (Sellers 1997). If the
radiological dose for the unmitigated case exceeds the guideline, mitigative or preventive safety
features, with appropriate safety-class and/or safety-significant functional classifications, are identified. The dose consequences are recalculated taking appropriate credit for the mitigating safety features to verify that the mitigated doses satisfy the guidelines. If there is a potential for a loss of nuclear criticality control, the engineered features designed to prevent a nuclear criticality are designated safety-class SSCs.

Where appropriate, based on the accident scenario and the safety features or functions required to prevent the occurrence of an accident, specific interlocks have been identified and assigned safety designations. These interlocks have been identified based on safety function or feature required and the specific interlock functions identified in Table A2-4.

Tables associated with each accident scenario provide the safety SSCs for the evaluated accidents and other defense-in-depth features. The tables may be used in cost–benefit analyses to evaluate any future improvements in any of these features or safety systems in order to improve operations, maintenance, or design by using the tables’ general arrangement of most important systems and features to least important.

A3.4.2 Design Basis Accident Analysis

This section presents a summary of the key assumptions and results of the DBA analyses that have been performed for the CSB. The DBAs are summarized based on the guidelines provided in DOE-STD-3009-94 and include the following categories:

- Mechanical damage of MCO (Section A3.4.2.1)
- Gaseous release from the MCO (Section A3.4.2.2)
- MCO internal hydrogen deflagration (Section A3.4.2.3)
- MCO external hydrogen deflagration (Section A3.4.2.4)
- Thermal runaway reactions inside the MCO (Section A3.4.2.5)
- Violations of design temperature criteria (Section A3.4.2.6).

The DBAs have been analyzed to quantify consequences and compare them with release limits for offsite consequences and evaluation guidelines for onsite consequences. The process is iterative, starting by taking no credit for mitigative features and comparing the results to the limits or guidelines. Credit is then taken for safety SSCs that prevent or mitigate the consequences to show that the results are below the release limits and evaluation guidelines. The process continues after the release limits and evaluation guidelines are met by identifying other SSCs that, while not designated as safety class or safety significant, provide additional mitigative features as defense in depth. In addition, each individual DBA section references a supporting calculational note that provides more detail.
All analyses have been performed based on the following MCO initial conditions defined in HNF-SD-SNF-TI-015:

1. Maximum particulate mass of aluminum hydroxide less than or equal to 9.47 kg in an MCO with bounding total particulate delivered to the CSB

   The bounding value is based on statistical analysis of good quality film thickness data at a 99% confidence level. To obtain this confidence level, a thickness of 1.8 times the actual mean of the measurements was used. This value was multiplied by the total assumed surface area and the theoretical density of aluminum hydroxide to obtain the bounding value. Additional information is documented in HNF-1527, *Estimates of Particulate Mass in Multi-Canister Overpacks*.

2. Maximum mass of uranium oxide hydrates less than or equal to 10.8 kg in an MCO delivered to the CSB

   Over 90% of this value is from sources that are quantified by analysis. The bounding values for these sources are based on fuel characterization data combined with very conservative assumptions of the amount of damaged fuel in the MCO and on literature data on hydrate decomposition. Less than 10% of the value is from canister particulate. To obtain this value, credit is taken for fuel cleaning, which must be validated by examination of a test batch of cleaned fuel assemblies and confirmed by periodic examination of assemblies during the process. Additional information on the quantity of hydrates in a bounding MCO is provided in HNF-1527 and in HNF-1523, *K Basin Particulate Water Content, Behavior, and Impact* (Rev. 1).

3. Maximum of two scrap baskets in an MCO delivered to the CSB

   This condition is controlled by TSR commitments in HNF-SD-WM-SAR-062, *K Basins Final Safety Analysis Report*. The safety of two scrap baskets has been demonstrated relative to all credible accident scenarios.

4. Maximum (not strongly adherent) UO₂ particulate less than or equal to 34 kg in an MCO after 40 years of storage at the CSB

   This is particulate that is generated after fuel cleaning. The requirement is met by bounding analysis that relies on TSR commitments in HNF-SD-WM-SAR-062 that limit the amount of scrap and damaged fuel in an MCO. This value is based on very conservative assumptions for both oxidation rates and elapsed time at various process steps. Additional information regarding the values for these process steps is documented in HNF-1527.
5. Maximum free water less than 200 g in an MCO delivered to the CSB

This value is based on processing steps at the conclusion of cold vacuum drying with time, pressure, and temperature conditions that ensure sufficient water evaporation and removal even with very conservative assumptions of fuel cracking and trapped particulate. Analysis confirming the feasibility of limiting free water and the associated processing requirements at the CVDF is provided in HNF-1851, *Cold Vacuum Drying Residual Free Water Test Description* (Rev. 2). This analysis is supported by whole-element characterization data and bounds the free water present in the MCO.

6. Leak rate of internal gas from MCO less than $10^{-5}$ standard cm$^3$/s

Before the MCO is shipped from the CVDF to the CSB, the MCO’s mechanical closure seal will have been demonstrated to meet a $10^{-5}$ standard cm$^3$/s total integrated leak test acceptance criterion as defined by ANSI N14.5-1987, *American National Standard for Radioactive Materials — Leakage Tests on Packages for Shipment*. The total integrated leakage criterion will be satisfied by a summation of test results for individual seals.

7. Maximum mass of uranium hydride (for heat and hydrogen generation) equal to 5.13 kg in an MCO delivered to the CSB and 9.08 kg after 40 years of storage at the CSB

Two methods were used to estimate bounding and nominal hydride inventories. The first method assumed particulate generated by in-basin corrosion is partially composed of uranium hydride, as found in literature experiments. The second method assumed an element contains uranium hydride equivalent to observations reported in single-element drying tests. Similar estimates were obtained from both methods, but the estimate based on element drying data was used to develop the bounding estimate of uranium hydride inventory. Additional information is provided in HNF-3372, *Uranium Hydride in Multi-Canister Overpacks*.

8. Maximum of 6,339 kg of uranium per MCO (based on five Mark IV fuel baskets)

The value is determined by the design of the MCO, which has room for five Mark IV fuel baskets. Each Mark IV fuel basket has positions for 54 fuel assemblies, and the maximum uranium mass in a Mark IV fuel assembly is 23.48 kg. The uranium mass in various fuel assemblies and the possible loading combinations are described in HNF-SD-SNF-TI-009, *105-K Basin Material Design Basis Feed Description for Spent Nuclear Fuel Project Facilities*.
Minimum inert gas pressure of 1.5 atm at 25 °C in an MCO delivered to the CSB.

Before the MCO is shipped from the CVDF to the CSB, it will have been chilled in accordance with SNF-2356, Spent Nuclear Fuel Project Cold Vacuum Drying Facility Operations Manual, and backfilled with helium to between 1.65 atm and 1.85 atm absolute.

Combustible loading controls are required to be in place to preserve the combustible loadings assumed in the fire hazard analysis (HNF-SD-SNF-FHA-002). The fire hazard analysis was prepared in accordance with DOE Order 5480.7A, Fire Protection, and HNF-PRO-350, Fire Hazard Analysis Requirements. The analysis evaluated potential fire risks at the CSB to determine whether any undue hazards required mitigation to protect workers, property, the public, or the environment. The conclusion of the analysis is that the fire risks present in the facility are being mitigated through engineering design and administration of control of fuel packages (HNF-SD-SNF-FHA-002). This conclusion is consistent with the results of the accident hazard analysis prepared for this chapter (HNF-SD-SNF-HIE-001). Consequently, the accident analysis in this chapter has only evaluated the potential for the DBAs associated with deflagrations caused by hydrogen generation resulting from radiolysis and fuel corrosion. The fire hazard analysis (HNF-SD-SNF-FHA-002) and its implementation plan define the facility requirements related to combustible loadings and necessitate TSR controls.

A3.4.2.1 Mechanical Damage of Multi-Canister Overpack. An MCO or the cask-MCO combination subjected to an accidental drop, impact, or shear force of sufficient magnitude could be damaged such that the MCO is breached or its internal geometry is substantially compromised. Several potential accidents at the CSB that could damage the MCO with mechanical forces are identified in HNF-SD-SNF-HIE-001 and are evaluated in SNF-3328. Three classifications of accidents have been selected for further evaluation: drop of the MCO or cask-MCO, shear of the MCO by the MHM, and impacts to the MCO other than drops and shears.

Drop of a Multi-Canister Overpack. An MCO or the cask-MCO combination subjected to an accidental drop of potentially sufficient magnitude could be damaged such that the MCO is breached or its internal geometry is substantially compromised. A number of potential drop accidents at the CSB have been identified for further discussion or evaluation. These accidents are listed in Section A3.4.2.1.1. Required controls for each accident are summarized in Section A3.4.2.1.5.

Drops that could lead to a ‘stuck’ MCO (with no loss of confinement or geometry) are all category 4 events as defined in NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36, and are therefore properly classified as accidents. The CSB has been designed to allow ready retrieval of SNF during normal and off-normal operation. Therefore, the CSB complies with the retrievability requirement for normal and off-normal conditions.
Having many different drop scenarios with different expected frequencies and consequences makes it difficult to plan detailed recovery actions for every eventuality. Therefore, a general recovery plan that maps out general approaches and thought processes will be used as guidance to develop a detailed, accident-specific recovery plan in the unlikely event an MCO is dropped.

Shear of a Multi-Canister Overpack. An MCO or the cask-MCO combination subjected to a shear force of sufficient magnitude could be damaged such that the MCO is breached or its internal geometry is substantially compromised. Several potential shear accidents at the CSB that could damage the MCO have been identified for further discussion or evaluation. These accidents are listed in Section A3.4.2.1.1.

Impacts to a Multi-Canister Overpack Other than Shears and Drops. An MCO or the cask-MCO combination subjected to a force of sufficient magnitude could be damaged such that the MCO is breached or its internal geometry is substantially compromised. Several accidents at the CSB that could potentially damage the MCO with mechanical forces have been identified. These accidents are listed in Section A3.4.2.1.1.

A3.4.2.1.1 Scenario Development.

Drop of a Multi-Canister Overpack. At the CSB the MCO is moved by both the receiving crane and the MHM. The MCO could be dropped, either when it is inside the transportation cask or when it is outside the cask. Several different scenarios that could cause an MCO drop have been considered. All of these drop scenarios are considered potentially serious hazards to the MCO in the CSB hazard analysis (HNF-SD-SNF-HIE-001):

D1 Cask-MCO drop from receiving crane onto receiving area
D2 Cask-MCO drop from receiving crane into cask receiving pit
D3 Cask-MCO drop from receiving crane into Fast Flux Test Facility (FFTF) pit or MHM maintenance pit
D4 Drop of MCO by MHM onto edge of cask receiving pit
D5 Drop of MCO by MHM back into cask
D6 Drop of MCO within MHM turret onto MHM turret deck
D7 Drop of MCO from MHM onto CSB operating area floor or onto storage tube covers of vault 2 or vault 3 in CSB
D8 Drop of MCO from MHM onto or into MHM maintenance pit or tube plug exchange facility
D9 Drop of MCO from MHM onto or into FFTF pit
D10 Drop of MCO by MHM into storage tube or sampling/weld station
D11 Drop of MCO by MHM onto another MCO located in the bottom of a storage tube
(no intermediate impact absorber involved)
D12 Drop of MCO by MHM onto storage tube plug because of inadvertent turret rotation
D13 Drop of MCO by MHM onto edge of storage tube
D14 Drop of MCO by MHM onto edge of sampling/weld station.

D1 — Cask-Multi-Canister Overpack Drop from Receiving Crane onto Receiving Area. The receiving crane is used to lift the cask-MCO from the transport trailer using a yoke. The crane is then translated to place the cask-MCO above the cask receiving pit, and the cask-MCO is lowered into the pit. The cask-MCO could be dropped from the receiving crane at any location through which the crane travels with the suspended cask-MCO. The CSB receiving crane has been designated a safety-significant and important-to-safety piece of equipment. The crane is not single-failure-proof but is rated for twice the load required for a safety-significant hoist.

Cask-MCO drops from the receiving crane could occur because of a failure in the lift system (e.g., the hook, hoist rope, cask yoke) or as a result of improper connection of the load to the hoist. Because the crane is designed to select criteria of ASME NOG-1, Type I, the crane load remains suspended during and after a design basis earthquake (DBE) event. The following five scenarios involving 60-in. drops from the receiving crane have been analyzed in WMTS-RPT-037, Canister Storage Building Multi-Canister Overpack Cask and Multi-Canister Overpack Impact Analyses.

1. **Cask-MCO Slapdown Drop onto the Concrete Deck.** In this case, two drop orientations of the cask-MCO are evaluated: a center-of-gravity over corner orientation and a shallow angle (15°) orientation. In both orientations, the cask-MCO is assumed to be suspended initially from the crane with the impacting corner of the cask-MCO 60 in. from the floor. The rigging then fails in a manner such that the cask-MCO is subjected to an oblique drop and slapdown onto the concrete deck of the CSB. For both orientations, the primary impacts are simulated as corner drops and the slapdowns are simulated separately as flat horizontal drops from heights that account for the additional rotational velocity of the cask center of gravity at impact.

2. **Cask-MCO End (Vertical) Drop onto the Concrete Deck.** In this case, the cask-MCO is assumed to be suspended initially from the crane with the bottom of
the cask-MCO 60 in. from the floor. The rigging then fails in a manner such that the
cask-MCO is subjected to a flat bottom-end drop onto the concrete deck and does
not tip over.

3. Cask-MCO Drop onto Edge of Cask Receiving Pit with Slapdown. This case is the
same as case 1 except the initial oblique drop is on the edge of the cask receiving pit
with slapdown (both shallow angle and center-of-gravity over corner orientations).
This slapdown is simulated by a horizontal side drop from an increased height that
accounts for the rotational velocity of the center of gravity of the cask.

4. Cask-MCO Drop Spanning Cask Receiving Pit. In this case, the cask-MCO is
assumed to drop from a height of 60 in. onto the cask receiving pit. The drop is
assumed to be a horizontal side drop with the center of gravity of the cask over the
geometric center of the pit.

5. Shallow Angle Cask-MCO Drop into the Cask Receiving Pit with Cask Impacting
on Opposite Edge of Vault Pit. In this case, the cask-MCO is assumed to drop into
the pit at a shallow angle off the horizontal (approximately 15°). Initially the
cask-MCO is assumed to be suspended from the crane with the impacting corner of
the cask-MCO 60 in. from the floor. The rigging then fails in a manner such that the
cask-MCO is subjected to a shallow angle drop into the pit and subsequent
slapdown onto the concrete deck near the edge of the vault pit. The primary impact
is simulated as a shallow angle drop impacting the opposite edges of the pit and the
slapdown is simulated separately as flat horizontal drops from heights that account
for the rotational velocity of the cask center of gravity at impact. The slapdown
sequence is that the cask initially impacts on its bottom corner in the pit and the top
end closure slaps down on the vault pit edge.

Using NRC and commercially accepted practices, the cask in these cases is considered to
be a rigid body impacting a yielding surface. The concrete and soil are considered real surfaces
that absorb the energy of impact. This is predicated on the shipping cask being much harder and
stiffer than a concrete surface. Consequently, the concrete and underlying soil act to decelerate
the cask.

Numerical simulations using the ABAQUS/Explicit computer program were used to
develop the average inertial load factors and inertial load factor time histories for the various
scenarios involving cask-MCO drops onto concrete surfaces at the CSB. Since the cask has a
natural frequency in excess of 33 Hz, the cask was modeled as a rigid body in the numerical
impact simulations. The impacted CSB concrete surfaces and soil subgrades were modeled as
real targets. The slope of the velocity time history profiles was used to determine the average
inertial load factors.

Quasi-static methods were used for structural evaluation of the cask-MCO closure. This
quasi-static method is based on D’Alembert’s principle of substituting an equivalent static force
for the inertial force created by impact. These inertial forces represented in the calculation as
equivalent static loads were determined by multiplying the weight of the cask by the average
inertial load factor of the cask. In addition, this average inertial load factor was multiplied by the
maximum dynamic load factor to account for the dynamic response of the cask.

Given the robustness of the cask and cask closure lid, the most vulnerable components of
the cask relative to loss of confinement during and after a free drop impact are the closure lid
bolts. The bolts were evaluated by classical linear elastic methods by the evaluation approach
specified in NUREG/CR-6007, Stress Analysis of Closure Bolts for Shipping Casks.
For conservatism, the closure system was considered a protected lid with the weight of the
closure lid applying a shear load to the bolts at the threads.

The loading evaluation for these cases (WMTS-RPT-037) shows that stresses in the bolts
are below the allowable limits specified in NUREG/CR-6007 for accident conditions. This is
demonstrated by the positive margins of safety based on the allowable stress intensity limits in all
cases. Results show that margins of safety on the reduced shank of the bolt are positive and the
margins of safety of the shear ring are large. During a flat bottom end drop, the only loading on
the closure lid bolts is the fixed edge moments and forces applied by the closure lid inertia. This
result in an average tensile load on the bolt that, when compared with the allowable stress
intensity limits, results in a margin of safety of 7.38. Initial oblique bottom impact loads on the
bolts were treated as axial and radial vector components of the impact force. In these cases, the
inertial load factors were relatively low compared with the horizontal side drop and bottom end
drop. As a result, the margins of safety on the closure bolts are large. Consequently, as
demonstrated by the positive margins of safety, the cask closure lid bolts, reduced bolt shank, and
shear ring remain elastic and maintain confinement in the various scenarios involving a drop of the
cask-MCO onto the CSB floor and floor structures.

To be conservative, a breach of the MCO as well as a breach of the cask following this
drop is assumed. Unmitigated, this event has been determined to be in the unlikely category
(SNF-4042). The gas leaving the MCO contains considerable hydrogen so a flammable mixture
of hydrogen, helium, and air forms in the space between the outside of the MCO and the inside of
the cask. This flammable mixture ignites, but the resulting pressure increase is not expected to
damage the cask (see Section A3.4.2.4).

D2 — Cask-Multi-Canister Overpack Drop from Receiving Crane into Cask
Receiving Pit. The cask receiving pit is fitted with an impact absorber designed to control the
dynamics of dropping an MCO cask containing an MCO into an empty cask (i.e., to limit the
deceleration). The impact absorber mitigates the damage to the MCO following the drop
accident. The impact absorber is required to limit the deceleration of a maximum weight MCO
dropped from the maximum height to less than 35 g. The MCO and internal baskets are designed
to maintain confinement under drops with decelerations limited below 35 g. The impact absorber
for the cask receiving pit is designed for a drop height of 281 in. The maximum height of the cask
receiving pit impact absorber is 45 in. The impact absorber design was developed through a
combination of both calculations and static and dynamic tests.
To be conservative, a breach of the MCO as well as a breach of the cask following this
drop is assumed. Unmitigated, this event has been determined to be in the unlikely category
(SNF-4042). Full-scale prototypic tests of the impact absorber were performed and are
documented in TR-003, Test Report for the CSB Prototypic Impact Absorbers. The tests used
video records to demonstrate acceptable deceleration. While a direct crush measurement of the
impact absorber indicated a deceleration load that was higher than expected and higher than the
design criteria, this was attributed to energy transferred to the ground upon impact.

HNF-SD-SNF-DR-003, Multi-Canister Overpack Design Report, demonstrates that the
MCO will meet design criteria. The design calculation shows that the Mark IA MCO’s internals
can withstand the design basis conditions of a 35 g vertical drop or a 101 g horizontal drop. The
MCO cask containing an MCO will not be damaged following a drop into the cask receiving pit.

D3 — Cask—Multi-Canister Overpack Drop from Receiving Crane into Fast Flux
Test Facility Pit or Multi-Canister Overpack Handling Machine Maintenance Pit. When
the receiving crane is moving the cask—MCO from the transport trailer to the cask receiving pit,
the cask—MCO is not to be moved over the FFTF or maintenance pits. The receiving crane
resolver tracks the east–west position of the receiving crane. Before the receiving crane reaches
the west side of the FFTF pit, the interlock removes power from the drive motors. Power may
only be restored to the crane by both the use of a supervisor-controlled fortress key at a remote
station and the operator depressing an override button on the crane. While this interlock does not
reduce the drop frequency, it does reduce the frequency of drops into the FFTF or maintenance
pits.

However, given certain errors, the cask—MCO could mistakenly be moved over the FFTF
or maintenance pits where it would be vulnerable to a potential drop into the pit. The
consequences of a cask—MCO drop by the receiving crane (from a height greater than 60 in.) into
the FFTF or maintenance pit with a slapdown have not been determined. While definitive
supporting calculations do not exist, no radiological releases or associated onsite or offsite dose
consequences are expected from such an accident because of the structural protection provided by
the cask. A breach of the MCO as well as a breach of the cask following this drop is assumed.
Unmitigated, this event has been determined to be in the unlikely category (SNF-4042).
Radiological releases are presented in Section A3.4.2.1.3. The gas leaving the MCO contains
considerable hydrogen so a flammable mixture of hydrogen, helium, and air forms in the space
between the outside of the MCO and the inside of the cask. This flammable mixture ignites, but
the resulting pressure increase is not expected to damage the cask (see Section A3.4.2.4).

D4 — Drop of a Multi-Canister Overpack by the Multi-Canister Overpack Handling
Machine onto Edge of the Cask Receiving Pit. If the MHM were to drop an MCO, the MCO
would impact the slope of the shield hatch and MCO guide assembly because of the shield hatch
and MCO guide assembly design and the centering guide installed in the MHM. The shield hatch
has a 75° incline, which would greatly reduce the impact force. The centering guide would limit
MCO tilt, which would ensure the impact area was always on the shield hatch incline.
An engineering analysis using the ABAQUS/Explicit computer code has been performed to address dropping an MCO onto the edge of the cask receiving pit (eccentric drop); the analysis is documented in SNF-5204, *Analysis for Eccentric Multi-Canister Overpack Drops at the Canister Storage Building (CSB-S-0073)*. This analysis predicted accelerations, plastic strain, and stress distributions, and identifies failure modes. The structural numerical models addressed nonlinear material behavior as well as impact dynamics. The weight of the MCO used in the analysis was 20,000 lb, and the drop height was conservatively assumed to be 90 in. The maximum equivalent MCO strain was found to be 18% and occurred on the MCO bottom near the point of impact. Since the failure strain is above 80%, it is not likely that through-the-wall cracking would occur although removal of local surface metal may occur. The maximum predicted equivalent plastic strain at the MCO lower weld was 0.4% during a drop into the standard storage tube. This is well below a 13% failure strain. The maximum tensile principal strain was in a radial direction, so any failure would be by spalling or flaking rather than through-the-wall cracking.

The deceleration of the MCO baskets was shown to be about 20 g or less, well within the design allowable of 35 g.

To be conservative, a breach of the MCO following this drop is assumed. Unmitigated, this event has been determined to be in the unlikely category (SNF-4042).

**D5 — Drop of Multi-Canister Overpack by the Multi-Canister Overpack Handling Machine Back into Cask.** An engineering analysis has been performed to address dropping an MCO back into the cask; the analysis is documented in SNF-5276, *Analysis for SNF MCO Drop into the Cask from the MHM with Air Cushion*. The MCO was assumed to drop from a height of 8.2 ft above the cask, enter the cask concentrically, and fall the additional 12.83 ft to the cask bottom. The MCO's fall would be slowed by air entrapment and the interface fit between the MCO and the cask. The shipping cask was assumed to be resting on the cask receiving impact absorber at the time of impact, and the energy absorbing properties of the cask receiving pit impact absorber were included in the analysis.

The engineering analysis (SNF-5276) showed that an MCO drop into the cask produces very large impact reactions on the MCO and its internals. Plastic strains were shown to occur in the bottom of the MCO, the sides of the MCO near the weld area, the basket support plates, the outside basket support posts, and the basket center support post. Large amounts of energy were absorbed during the impact by these components.

The maximum equivalent plastic strain in the bottom of the MCO was 2.0%. This is below the calculated effective failure strain of 15%. The effective failure strain value was calculated based on the multi-axial stress state (triaxiality factor), temperature, and strain rate. Breaching of the MCO pressure boundary by through-wall cracking is not expected.

Equivalent plastic strains in the weld area at the juncture of the MCO walls and the MCO bottom were 2.1%. This is below the calculated 12.5% effective failure strain. The maximum equivalent plastic strain occurred at the inside surface of the MCO in the weld area. Breaching of the MCO pressure boundary by through-wall cracking in the weld area is also not expected.
To be conservative, a breach of the MCO following this drop is assumed. Unmitigated this event has been determined to be in the unlikely category (SNF-4042).

**D6 — Drop of Multi-Canister Overpack within Multi-Canister Overpack Handling Machine Turret onto Multi-Canister Handling Machine Turret Deck.** When the MHM is traveling in the operating area with an MCO fully raised into the MCO cavity of the MHM turret, the MCO is vulnerable to a potential drop onto the MHM turret deck while partially rotated over the opening. Engineering analysis has been performed to evaluate an MCO drop onto the MHM turret deck (WMTS-RPT-037). As a worst case, it was assumed that the MCO dropped at a shallow angle of 1° off the vertical axis and onto a squared edge of the MHM turret base opening. The drop was assumed to result in an edge impact approximately one-eighth the diameter in from the edge of the MCO.

The results show that the most vulnerable areas of the MCO are not breached and the MCO maintains confinement even at a metal temperature of 270 °F. The maximum equivalent strain (which is a true strain) is 15.5% at the impact point on the MCO bottom plate outside surface. The maximum equivalent strain at the inside surface on the bottom-end-plate-to-shell weld joint is 3.72%. Based on the CSB Expert Panel criteria adjusted for true strain (SNF-5276), the failure strain value at 270 °F is 6.7%. The margin of safety at the impact point for the base metal is 2.1, and in the weld area, the margin of safety is 0.8 (SNF-5276). The margin of safety used is calculated as follows:

\[(\text{allowable value} / \text{calculated value}) - 1\]

The allowable values for the MCO are ASME Boiler and Pressure Vessel Code, Section III, class 1 allowables for Service Level D (ASME 1995).

The maximum strain energy density at the impact point on the MCO bottom plate outside surface is 7,317 in-lbf/in³, and the maximum strain energy density at the inside surface on the bottom-end-plate-to-shell weld joint is 1,258 in-lbf/in³. For Type 304L stainless steel at 270 °F, the minimum yield strength is 19.6 ksi, and the minimum ultimate strength is 62.5 ksi at 40% strain, equating to a minimum rupture toughness of 16,420 in-lbf/in³. Based on the material toughness criteria, the maximum allowable strain energy density of the base metal is 8,210 in-lbf/in³. The maximum allowable strain energy density in the weld region is 4,105 in-lbf/in³. The margin of safety of strain energy density is 0.12 at the impact point and 2.3 in the weld region.

The maximum equivalent plastic strain and strain energy density data show that at the contact locations, the outer surface of the bottom end closure of the MCO is dented. However, the data also show that this damage is only on the surface and does not extend into the inside surface of the bottom end closure plate. The maximum equivalent plastic strain at the weld joint between the MCO shell and bottom end closure plate is very low and well below the established limits for the material. In the weld joint region, the maximum equivalent plastic strain and strain energy density are on the inside surface of the confinement boundary.
To be conservative, a breach of the MCO following this drop is assumed. Unmitigated, this event has been determined to be in the unlikely category (SNF-4042).

D7 — Drop of Multi-Canister Overpack from Multi-Canister Overpack Handling Machine onto Canister Storage Building Operating Area Floor or onto Storage Tube Covers of Vault 2 or Vault 3 in Canister Storage Building. If errors are made in failing to replace the storage tube plug (or cover plug or center plate) when the MHM has finished raising an MCO into the MCO cavity in the MHM turret, and the MHM is inadvertently moved a short distance over the operating floor, the MCO is vulnerable to a potential drop onto the operating area floor from the fully raised position in the MCO cavity in the MHM turret when no credit is taken for the MHM interlocks. Anytime an MHM is carrying an MCO in the MCO cavity in the MHM turret and stops anywhere on the operating area, the MCO is vulnerable to a potential drop onto the operating area floor from the fully raised position in the MCO cavity in the MHM turret when the MHM is rotated to the MCO cavity of the MHM turret position, providing no credit is taken for the MHM interlocks. Whenever an MHM is carrying an MCO in the MCO cavity in the MHM turret and stops over vaults 2 or 3, the MCO is vulnerable to a potential drop onto the vault covers from the fully raised position in the MCO cavity in the MHM turret when the MHM is rotated to the MCO cavity in the MHM turret position, providing no credit is taken for the MHM interlocks. A breach of the MCO following this drop is assumed. Unmitigated, this event has been determined to be in the extremely unlikely category (SNF-4042).

D8 — Drop of Multi-Canister Overpack from Multi-Canister Overpack Handling Machine onto or into Multi-Canister Overpack Handling Machine Maintenance Pit or Tube Plug Exchange Facility. When the MHM is moving an MCO in the operating area, the MHM is not to be moved over the MHM maintenance pit. However, given certain errors, the MHM containing an MCO could mistakenly be moved over the MHM maintenance pit or tube plug exchange facility, where the MCO would be vulnerable to a potential drop into the pit providing no credit is taken for the MHM interlocks. A breach of the MCO following this drop is assumed. Unmitigated, this event has been determined to be in the extremely unlikely category (SNF-4042).

D9 — Drop of Multi-Canister Overpack from Multi-Canister Overpack Handling Machine onto or into Fast Flux Test Facility Pit. When the MHM is moving an MCO in the operating area, the MHM is not to be moved over the FFTF pit. However, given certain errors, the MHM containing an MCO could mistakenly be moved over the FFTF pit providing no credit is taken for the MHM interlocks. A breach of the MCO following this drop is assumed. Unmitigated, this event has been determined to be in the extremely unlikely category (SNF-4042).

D10 — Drop of Multi-Canister Overpack by Multi-Canister Overpack Handling Machine into Storage Tube or Sampling/Weld Station. Each storage tube bottom is fitted with a bottom impact absorber of stainless steel tubes that has been designed to control the dynamics of dropping a single MCO into an empty storage tube. The bottom impact absorber will mitigate damage to an MCO following such a drop. These bottom impact absorbers are required...
to limit the deceleration of a maximum weight MCO dropped from the maximum height to less than 35 g. The MCO and its internal baskets are designed to maintain confinement under drops with decelerations limited below 35 g as specified in HNF-S-0426, *Performance Specification for Spent Nuclear Fuel Project, Multi-Canister Overpack*. Therefore, the MCO will not be damaged following a drop into the storage tube. The adequacy of the bottom impact absorber has been demonstrated by a full-scale test. To be conservative, a breach of the MCO following this drop is assumed. Unmitigated, this event has been determined to be in the unlikely category (SNF-4042).

An intermediate impact absorber is placed between two MCOs stacked in a storage tube. The intermediate impact absorber is designed to mitigate damage to both MCOs if the top MCO is accidentally dropped during storage tube loading. These intermediate impact absorbers are required to limit the deceleration of an MCO dropped on top of another MCO to less than 35 g. The MCOs will not be damaged following a drop on top of another MCO in the storage tube. The adequacy of the intermediate impact absorber has been demonstrated by a full-scale test.

Each sampling/weld station pit is fitted with an impact absorber of stainless steel tubes that has been designed to control the dynamics of dropping an MCO into the sampling/weld station rotating shield. The sampling/weld station impact absorber will mitigate damage to the MCO after such a drop. These impact absorbers are required to limit the deceleration of a maximum weight MCO dropped from the maximum height to less than 35 g. The MCO and its internal baskets are designed to maintain confinement under drops with decelerations limited below 35 g (HNF-S-0426). Therefore, the MCO will not be damaged following a drop into the sampling/weld station rotating shield. The design adequacy of the impact absorber has been demonstrated by a full-scale drop test.

**D11 — Drop of Multi-Canister Overpack by Multi-Canister Overpack Handling Machine onto Another Multi-Canister Overpack Located in the Bottom of a Storage Tube (No Intermediate Impact Absorber Involved).** When the MHM is placing an MCO into a storage tube that contains one MCO but which has no intermediate impact absorber, each MCO would be vulnerable to a potential drop of the MCO hanging from the MHM.

A drop of an MCO onto another MCO already loaded into the storage tube would cause plastic deformation to the bottom of the dropped MCO and the top of the impacted shield plug, but the confinement functions of either MCO would not be jeopardized. The deceleration associated with such a drop would not damage the MCO internals. HNF-SD-SNF-DP-007, *Multi-Canister Overpack Analysis File Documentation*, Appendix D, "Multi-Canister Overpack to Multi-Canister Overpack Drop Analysis," documents the simulation of a 31-ft drop of a loaded MCO onto another MCO. A bottom impact absorber is in place under the first MCO, but no intermediate impact absorber is in place between them. The analysis indicates that plastic deformation of the impacted shield plug seal and shoulder would be a maximum of 0.030 in. The impact absorber under the bottom MCO would be compressed 6 to 9 in. To be conservative, a breach of the MCO following this drop is assumed. Unmitigated, this event has been determined to be in the unlikely category (SNF-4042).
D12 — Drop of Multi-Canister Overpack by the Multi-Canister Overpack Handling Machine onto the Storage Tube Plug because of Inadvertent Turret Rotation. Anytime an MHM is carrying an MCO in the MCO cavity in the MHM turret and stops over a storage tube plug, the MCO is vulnerable to a potential drop onto the storage tube plug pintle from the fully raised position in the MCO cavity in the MHM turret when the MHM is rotated to the MCO cavity in the MHM turret position providing no credit is taken for the MHM interlocks. A breach of the MCO following this drop is assumed. Unmitigated, this event has been determined to be in the extremely unlikely category (SNF-4042).

D13 — Drop of Multi-Canister Overpack by Multi-Canister Overpack Handling Machine onto Edge of Storage Tube. The storage tube is supported off the floor. The lower flange of the tube is 4.5 in. thick and has a 45° incline to reduce any direct impact between a dropped MCO and the edge of the storage tube following an eccentric drop event. An engineering analysis (SNF-5204) has been performed to address dropping an MCO onto the edge of a storage tube (eccentric drop). The drop was conservatively assumed to hit the lower flange of the storage tube (which has a 45° incline) not the guide funnel (which has a 75° incline). The weight of the MCO used in the analysis was 20,000 lb, and the drop height was conservatively assumed to be 90 in. The MCO was assumed to be tilted by no more than 2°21' from vertical because of the MCO centering guide installed in the MHM. If a drop tilted more than 2°21', the MCO would most likely hit the funnel placed above the upper flange. The funnel has a 75° entrance angle, and the impact would be less severe than an impact on the lower flange. The engineering analysis predicted accelerations, plastic strain, and stress distributions, and identified MCO failure modes. The structural numerical models address nonlinear material behavior as well as impact dynamics. The tube bellows, which has an insignificant effect, was not included in the model.

According to the analysis, the maximum equivalent MCO strain was 40% to 45% and occurred on the MCO bottom near the point of impact (SNF-5204). Since the failure strain is above 80%, it is not likely that through-the-wall cracking will occur although removal of local surface metal may occur. The maximum predicted equivalent plastic strain at the MCO lower weld was 2% during the drop into the standard storage tube. This is well below a 13% failure strain. The maximum tensile principal strain was in a radial direction, so any failure would be by spalling or flaking rather than through-the-wall cracking. The deceleration of the MCO baskets was shown to be about 20 g or less, well within the design allowable of 35 g. The deformations of the MCO and standard storage tube were less than the tube clearance, so no wedging of the MCO in the tube is expected. To be conservative, a breach of the MCO following this drop is assumed. Unmitigated, this event has been determined to be in the unlikely category (SNF-4042).

D14 — Drop of Multi-Canister Overpack by Multi-Canister Overpack Handling Machine onto Edge of a Sampling/Weld Station. The MCO impacts a 75° inclined surface on the shield hatch ring at the cask receiving pit. The impact at the sampling/weld station is on a similar 75° inclined surface on the shield halves. The shield halves at the sampling/weld station is less massive and held in place less securely than the shield hatch ring of the cask receiving pit and...
will cause less damage to the MCO. Therefore, the results of an MCO drop onto the edge of the
cask receiving pit bound those for a drop onto the edge of the sampling/weld station.

The MCO will impact the incline of the shield halves because of the design of the shield
halves and the MCO centering guide installed in the MHM. The shield halves have a 75° incline,
which will greatly reduce the impact force. The MCO centering guide will limit the MCO tilt,
which ensures that the impact area is always on the incline of the shield halves. To be
conservative, a breach of the MCO following this drop is assumed. Unmitigated, this event has
been determined to be in the unlikely category (SNF-4042).

Shear of a Multi-Canister Overpack. At the CSB the MCO is hoisted by both the
receiving crane and the MHM. The MCO could be sheared, either when it is inside the
transportation cask or when it is outside the cask. Several different scenarios that could cause an
MCO shear (which would be a localized breach or tear and not complete severance of the MCO)
have been considered. All of these shear scenarios are considered potentially serious hazards to
the MCO in the CSB hazard analysis (HNF-SD-SNF-HIE-001):

S1  Shear of cask–MCO by moving receiving crane while cask–MCO is partially
     lowered into cask receiving pit, or shear of cask–MCO because of collision between
     MHM and receiving crane as receiving crane lowers cask–MCO into cask receiving
     pit

S2  Shear of MCO by MHM turret rotation with MCO partially retrieved into MHM at
cask receiving pit, storage tube, or sampling/weld station

S3  Shear of MCO by MHM translational movement with MCO only partially retrieved
     into MHM at cask receiving pit

S4  Shear of MCO while MCO is partially retrieved from or inserted into cask receiving
     pit, storage tube, or sampling/weld station during significant seismic event

S5  Shear of MCO by MHM translational movement with MCO only partially retrieved
     into MHM turret at storage tube or sampling/weld station.

S1 — Shear of Cask–Multi-Canister Overpack by Moving Receiving Crane while
Cask–Multi-Canister Overpack is Partially Lowered into Cask Receiving Pit, or Shear of
Cask–Multi-Canister Overpack because of Collision between Multi-Canister Overpack
Handling Machine and Receiving Crane as Receiving Crane Lowers Cask–Multi-Canister
Overpack into Cask Receiving Pit. When the receiving crane is lowering the cask–MCO into
the cask receiving pit and the cask–MCO is partially inserted into the cask receiving pit, the
cask–MCO is vulnerable to a potential shear of the cask between the pit wall and the cask wall if
the receiving crane were to be moved at this time. The shear forces from accidental movement of
the receiving crane are bounded by those from collision of the MHM with the receiving crane and
are therefore insufficient to breach the cask–MCO. No breach of the MCO following this
collision is predicted by analysis. Unmitigated, this event has been determined to be in the anticipated category (SNF-4042).

When the receiving crane is lowering the cask–MCO into the cask receiving pit and the cask–MCO is partially inserted into the cask receiving pit, design features preclude the MHM from colliding with the crane. It is conservative to assume that the cask–MCO is vulnerable to a potential shear of the cask from the MHM colliding with the receiving crane and subsequently with the cask; the cask provides confinement. Analysis documented in FDT-137, *MHM Impact with Cask/MCO during Insertion into CSB Transfer Pit* (Petersen 1998b), has shown that a collision would not breach the cask–MCO. MHM interlock P5, which inhibits receiving crane and MHM interaction, was not credited in the frequency calculation (SNF-4042); it is considered part of the defense-in-depth measures.

S2 — Shear of Multi-Canister Overpack by Multi-Canister Overpack Handling Machine Turret Rotation with Multi-Canister Overpack Partially Retrieved into Multi-Canister Overpack Handling Machine at Cask Receiving Pit, Storage Tube, or Sampling/Weld Station. When the MHM is raising the MCO into the MHM turret and the MCO is partially removed from the transportation cask, the storage tube, or sampling/weld station, the MCO is vulnerable to a potential shear from premature MHM turret rotation.

Analysis documented in SNF-5930, *Structural Analysis of Multi-Canister Overpack for Accidental Movement of Multi-Canister Overpack Handling Machine during Multi-Canister Overpack Lifting Operations*, demonstrates that a shear of the MCO is not possible using the turret rotation motor. For a turret rotational speed of 2 rpm, the kinetic energy is 1,300 ft-lbf. A locked rotor torque of 63 lb-ft results in a force on the MCO of 10,000 lbf. This force applied over a distance of 0.2 in. (penetration to fracture) increases the total energy to 1,470 ft-lbf. The energy required to initiate puncture of the MCO shell is 3,000 ft-lbf; therefore, the shell will not puncture from the inertial effects of the rotating turret combined with the drive motor effects. The peak force required to initiate puncture is 270,000 lbf. This value is roughly 27 times the 10,000 lbf the motor and gearing are capable of applying to the MCO at a locked rotor torque of 63 lb-ft. No breach of the MCO following turret rotation is predicted by analysis. Unmitigated, this event has been determined to be in the anticipated category (SNF-4042).

S3 — Shear of Multi-Canister Overpack by Multi-Canister Overpack Handling Machine Translational Movement with Multi-Canister Overpack only Partially Retrieved into Multi-Canister Handling Machine at Cask Receiving Pit. When the MHM is raising the MCO into the MHM turret and the MCO is partially removed from the transportation cask, the MCO is vulnerable to a potential shear from premature MHM translational motion. A breach of the MCO following the shear forces from MHM translational movement is assumed. Unmitigated, this event has been determined to be in the unlikely category (SNF-4042). Analysis documented in SNF-5930 shows that the internal geometry of the Mark IA baskets remain intact (no rearrangement of internal geometry).
S4 — Shear of Multi-Canister Overpack while Multi-Canister Overpack is Partially Retrieved from or Inserted into Cask Receiving Pit, Storage Tube, or Sampling/Weld Station during Significant Seismic Event. When the MHM is raising the MCO into the MHM turret and the MCO is partially removed from the shipping cask, the MCO is vulnerable to a potential shear from a seismic event. This shear event includes the potential shear caused by turret rotation or the potential shear caused by translational movement of the MHM. This shear event is prevented by the bridge and trolley seismic restraints as well as by the turret locking pin and base locking pin. Calculations have been performed to demonstrate the design adequacy of the seismic restraints and locking pins for seismic events with forces up to those of the DBE. To be conservative, a breach of the MCO following the shear forces from MHM translational movement during a significant seismic event is assumed. Unmitigated, this event has been determined to be in the unlikely category (SNF-4042).

S5 — Shear of Multi-Canister Overpack by Multi-Canister Overpack Handling Machine Translational Movement with Multi-Canister Overpack only Partially Retrieved into Multi-Canister Overpack Handling Machine Turret at Storage Tube or Sampling/Weld Station. When the MHM is raising the MCO into or lowering the MCO from the MHM turret and the MCO is partially removed from the storage tube or sampling/weld station pit, the MCO is vulnerable to a potential shear from premature MHM translational motion.

Analysis shows that the internal geometry of the Mark IA baskets remain intact (SNF-5930). To be conservative, a breach of the MCO following the shear forces from MHM translational movement is assumed. Unmitigated, this event has been determined to be in the unlikely category (SNF-4042).

Impacts Other than Shears and Drops of a Multi-Canister Overpack. At the CSB impacts to the MCO or to the cask–MCO could also be caused by events other than shears and drops. Several different scenarios that could cause an impact to the MCO have been considered. All of these impact scenarios are considered potentially serious hazards to the MCO in the CSB hazard analysis (HNF-SD-SNF-HIE-001):

O1 Drop of receiving crane yoke onto cask lid, or drop of cask lid onto MCO

O2 Collapse of CSB structure and impact to MCO during seismic event

O3 Drop of cask receiving pit shield hatch and MCO guide assembly’s shield hatch plate by MHM onto MCO, or drop of storage tube plug or sampling/weld station enter shield plate by MHM onto MCO in storage tube or sampling/weld station

O4 MHM fall or damage caused by a seismic event

O5 Drop of intermediate impact absorber onto MCO.
O1 — Drop of Receiving Crane Yoke onto Cask Lid, or Drop of Cask Lid onto Multi-Canister Overpack. When the receiving crane is raising the cask-MCO yoke away from the cask-MCO in the cask receiving pit, the cask lid is vulnerable to a potential drop of the yoke. In the case of the lifting yoke and crane rigging falling onto the MCO cask lid, the acceptable performance criterion is the cask lid not being punctured. Acceptable performance of the MCO cask lid is demonstrated by a quantitative comparative analysis based on a cask puncture test (WMTS-RPT-037).

For this scenario, the impact energy of the lifting yoke, crane rigging, and hook dropping onto the cask-MCO lid was compared with the cask-MCO puncture resistance analysis documented in Design Analysis Report for the TN-WHC Cask and Transportation System (TN 1996). The following assumptions were made in the comparative energy evaluation:

- Crane two-blocking causes the lifting yoke and crane hook and rigging to drop onto the cask-MCO lid while the cask is in the cask receiving pit.
- The total weight of the lifting yoke and crane hook and rigging is 6,200 lb.
- The cask-MCO is elevated to a working height, therefore the closure bolts are at CSB floor level.
- For conservatism, the protection provided by the lifting brackets and trunnions is not considered.
- The crane hook impacts the cask lid.

Based on the above assumptions, the drop height of the lifting yoke, crane hook, and rigging is 28 ft. The top of the cask closure lid, excluding the trunnion brackets, is 11 in. off the floor. This results in a drop height of 27 ft, 1 in. onto the closure lid. The potential energy from such a drop is 168 kip-ft.

The top end cask puncture evaluation (TN 1996) was performed in accordance with Title 10, Code of Federal Regulations, Part 71, “Packaging and Transportation of Radioactive Materials,” Section 71.73, “Hypothetical Accident Conditions” (10 CFR 71). The evaluation (TN 1996) determined that the loaded cask-MCO could survive a 40-in. drop onto a 6-in. solid vertical cylinder (puncture bar) mounted on an unyielding surface.

The impact energy of the lifting yoke and crane hook and rigging was calculated as being approximately 19% less than the impact energy evaluated in the Transnuclear (TN 1996) design analysis report. Therefore, there is margin of safety of 19% based on the cask puncture bar test. This is in addition to cask-MCO margin of safety in resisting puncture of 6.07 shown in the Transnuclear (TN 1996) evaluation.
The tent gantry hoist in the load-in/load-out area will be used to remove the cask lid from the cask-MCO. When the tent gantry hoist is raising the cask lid away from the cask-MCO in the cask receiving pit, the MCO is vulnerable to a potential drop of the cask lid. An analysis has been performed to evaluate cask lid drops (HNF-SD-SNF-DP-007). The analysis concluded that the MCO maintains confinement of all SNF during and after a 1.5-m (approximately 60 in.) drop of a cask lid. A drop of the cask lid from the tent gantry hoist onto the MCO would cause only local cosmetic damage at the point of impact, but the MCO would maintain its confinement functions. Local plastic damage would only be found by close inspection of the hardware after the event. If the receiving crane were used to remove the cask lid from the cask, the reliability of the receiving crane makes the consequences of the cask lid drop acceptable for the likelihood of a receiving crane drop. No breach of the MCO following drop of the cask lid by the tent gantry hoist is predicted by analysis. A breach of the MCO following drop of the cask lid by the receiving crane is assumed. Unmitigated, this breach event has been determined to be in the unlikely category (SNF-4042).

O2 — Collapse of Canister Storage Building Structure and Impact to Multi-Canister Overpack during Seismic Event. Given a seismic event occurs, there is potential for the CSB facility to structurally fail and fall to the operating area deck and to strike an MCO exposed at the cask receiving pit or the sampling/weld station pit. The CSB has been designed to withstand the DBE.

O3 — Drop of Cask Receiving Pit Shield Hatch and MCO Guide Assembly’s Shield Hatch Plate by Multi-Canister Overpack Handling Machine onto Multi-Canister Overpack, or Drop of Storage Tube Plug or Sampling/Weld Station Center Shield Plate by Multi-Canister Overpack Handling Machine onto Multi-Canister Overpack in Storage Tube or Sampling/Weld Station. When the MHM is raising the cask receiving pit shield hatch and MCO guide assembly’s shield hatch plate, it is possible that the MHM could drop the shield hatch plate toward the MCO in the cask. However, it is physically impossible for a dropped shield hatch plate to make contact with the top of an MCO because of the dimensions of the shield hatch plate and depth of the top of the MCO below the shield hatch and MCO guide assembly.

When the MHM is raising or lowering a storage tube plug or sampling/weld station center shield plate with the tube plug grapple, it is possible that the MHM could drop the storage tube plug or sampling/weld station center shield plate toward the MCO in the storage tube or the sampling/weld station pit. However, it is physically impossible for a dropped tube plug or cover plug to make contact with the top of an MCO because of the dimensions of the plug and depth of the top of the MCO below the storage tube embed or the sampling/weld station shielding.

O4 — Multi-Canister Overpack Handling Machine Fall or Damage Caused by a Seismic Event. Evaluations documented in SNF-5984, Multi-Canister Overpack Handling Machine Trolley Seismic Uplift Constraint Design Loads, were performed to study the behavior of the MHM during a DBE. These evaluations considered the effects on the MHM both with and without the seismic restraints and without the rail stops. All cases studied demonstrate that the
MHM will not tip over or collapse during a DBE. For the unrestrained seismic event, the MHM trolley seismic uplift restraints exceed the ASME-NOG-1 allowables. The design of the MHM will be strengthened to meet ASME-NOG-1 requirements, and this modification will be incorporated before CSB FSAR implementation. A potential MHM tip-over caused by the rail stops (if the MHM is near the rail stops without seismic restraints) during a DBE was not analyzed. The postulated scenario was judged to have very low likelihood because the MHM is rarely located close to the rail stops without the seismic restraints applied.

**O5 — Drop of Intermediate Impact Absorber onto Multi-Canister Overpack.** The acceptable performance criterion for the intermediate impact absorber impacting the top of an MCO is that the critical confinement boundaries of the MCO remain below yield under the impact loading.

A comparative qualitative analysis of an intermediate impact absorber drop onto the top of an MCO was performed based on HNF-SD-SNF-DP-007, the impact energy, and the design and weight of the intermediate impact absorber (WMTS-RPT-037).

The evaluation demonstrates that the MCO after a free drop impact of the intermediate impact absorber onto the MCO maintains confinement. Analysis documented in HNF-SD-SNF-DP-007 shows the MCO maintains confinement after the impact from a 31-ft free fall of another fully loaded MCO onto the top of a stationary MCO in the CSB storage tube. It was assumed that the MCO canister cover was not installed and the MCO shield plug area was exposed to the impact (HNF-SD-SNF-DP-007). The comparative analysis shows that based on the difference in weight of the MCO (20,000 lb) and the intermediate impact absorber (630 lb), the impact energy of the intermediate impact absorber is significantly less than the impact of a fully loaded MCO (620 kip-ft versus 19.5 kip-ft) (WMTS-RPT-037). Consequently, the MCO, with or without the cover cap, will maintain confinement after a free fall drop of the intermediate impact absorber onto a stationary MCO in the CSB storage tube.

**A3.4.2.1.2 Source Term Analysis.** The quantity of particulate released following an MCO breach caused by mechanical damage depends on the initial aerosol concentration, which includes that generated by the mechanical accident forces, and subsequent particulate entrainment. The potential sources of particulate entrainment come from venting of pressurized powder and aerodynamic entrainment. Both of these sources represent the effects of local gas velocities to entrain particles. The key difference between two entrainment sources is that the powder venting source is intended to represent a situation in which the gas flow field is predominantly through the particle bed, while the aerodynamic entrainment source is intended to represent a situation in which the gas flow field is predominantly outside the bed.

The amount of material available for release depends on the amount of particulate mass generated in the MCO during processing. This quantity is a function of processing times and temperatures as well as of the condition of the fuel in the MCO (e.g., exposed surface area of fuel). HNF-SD-SNF-TI-015 provides two estimates of this quantity: one for the safety basis, which is considered the bounding value, and the other for the design basis, which is considered the
nominal value. When the MCO is received at the CSB, the safety basis value is 23.1 kg uranium
and the design value is 0.54 kg uranium (HNF-SD-SNF-TI-015). After 40 years of storage at the
CSB, the safety basis, or bounding, value is 30 kg uranium and the design basis, or nominal, value
is 1.85 kg uranium (HNF-SD-SNF-TI-015). Because of the conservative methodology used to
calculate the bounding value, 30 kg is considered representative of the 99th percentile.

The MCO blowdown scenario features films of particles, not deep powder beds. While it is
true that there is some gas flow through particle layers during a blowdown, experiments
documented in Technical Report 11.6: Resuspension of Deposited Aerosols Following Primary
System or Containment Failure (IDCOR 1984) clearly show that this is a trivial effect.
Experiments that underlie the powder venting source term feature a pressurized cylinder packed
with powder. During depressurization, the flow velocity is zero at the bottom of the particle bed
when there is no underlying gas plenum, and the velocity increases to its maximum at the exit,
which is at the top. The gas velocity at the top of the bed must be enough to fluidize the bed
locally and to entrain the particles. When the particle bed height was decreased in these
experiments, the velocity at the top of the bed decreased in linear proportion to the distance, and
there was a bed height at which little or no entrainment could occur. In such a situation, there
was gas flow through the bed during depressurization, but the local velocity was simply not
sufficient for entrainment (IDCOR 1984). Therefore, for application to the MCO blowdown
accident scenarios, in which particle layers are thin and relatively well-distributed throughout the
MCO, the pressurized powder venting source term is not applicable, and a source term based
upon aerodynamic entrainment should be used.

DOE-HDBK-3010-94 documents ARRs and RFs. Based on the data reported in
DOE-HDBK-3010-94, a bounding (95th percentile) ARR of $4 \times 10^4$/h and an RF of 1.0 were
selected. The contents of the MCO are intact fuel elements tightly packed in fuel baskets and
pieces of fuel elements housed in scrap baskets. Particulate matter swept upward by streams of
flowing gas within the MCO must take a tortuous path through the MCO to exit the MCO. For
shielded powder, where the aerodynamic stresses are reduced by debris or exposure to static
conditions, DOE-HDBK-3010-94 recommends an ARR of $4 \times 10^4$/h and an RF of 0.2. These
values were selected as nominal (50th percentile).

The duration of the blowdown depends on the flow area of the leak and the initial MCO
pressure. The MCO depressurizes to atmospheric pressure in less than 1 minute, even for an
MCO pressure of about 150 lb/in² gauge and a 0.25-in. hole. The MCO depressurizes in a much
shorter time for lower pressures and larger leak paths. For the calculation of the source term, a
nominal (50th percentile) value of 10 seconds and a bounding (95th percentile) time of 60 seconds
were selected.

Using the methodology described in Section A3.4.1.3 and the bounding and nominal values
identified above, the bounding (95th percentile) value for the material released through
aerodynamic entrainment is calculated to be $2.0 \times 10^4$ g, as shown in Table A3-8.
The aerosol concentration inside the MCO at the time of the release depends on prior MCO handling conditions and the mechanical forces exerted on the MCO structure by the accident. The powder in the MCO at rest could be ejected into the gas volume by the response of the underlying solid MCO and fuel substrate to vibration or jolting induced by impact or falling debris. According to DOE-HDBK-3010-94, the value of the ARF under such circumstances should exceed the value of the ARF for aerodynamic suspension alone but be less than the ARF for the free fall of powder. The powder undergoing vibration shock is bounced into the gas while subject to the same gas velocities as those for aerodynamic entrainment. Based on the discussion in DOE-HDBK-3010-94, a bounding (95th percentile) ARF of $1 \times 10^{-3}$ and an RF of 1.0 were chosen for the suspension of powder-like surface contamination by shock vibration. A nominal (50th percentile) value of $4 \times 10^{-4}$ was chosen for the ARF and a nominal (50th percentile) value of 0.2 was chosen for the RF based on evaluation of the discussion in DOE-HDBK-3010-94. The pressure in the MCO at the start of the blowdown determines the fraction of the suspended particulate that is released. The initial pressure varies from 1.50 atm (corresponding to the helium fill pressure of 22 lb/in² absolute at the CVDF) to a maximum pressure of 5.2 atm (as defined in
the Technical Databook [HNF-SD-SNF-TI-015]). To allow for some additional margin in the
calculation, a pressure of 2.0 atm is considered nominal (50th percentile) and corresponds to an
LPF of 0.5, while a pressure of 6.0 atm is considered bounding (95th percentile) and corresponds
to an LPF of 0.83.

Using the methodology described in Section A3.4.1.3 and the bounding and nominal values
identified above, the bounding (95th percentile) value for the material released by impact is
calculated to be 1.2 g, as seen in Table A3-9.

| Table A3-9. Initial Aerosol Concentration Source Term for
| Mechanical Damage of the Multi-Canister Overpack. |
| Nominal | Bounding | EF | Percentile | SNV | ln(EF)/SNV |
| MAR     | 1.85 kg  | 30 kg | 16.22 | 99 | 2.326 | 1.198 |
| ARF     | 4.00 E-04 | 1.00 E-03 | 2.50 | 95 | 1.645 | 0.557 |
| RF      | 0.20     | 1.00  | 5.00  | 95 | 1.645 | 0.978 |
| LPF_{MCO} | 0.50   | 0.83  | 1.67  | 95 | 1.645 | 0.311 |

Note: The 1.645 is the standard normal variable corresponding to the 95% upper confidence limit. For the 99% upper
confidence limit, the corresponding value is 2.326.

\*LPF_{MCO} = (MCO pressure - 1 atm) / (MCO pressure). At a nominal MCO pressure of 2.0 atm, the LPF_{MCO} is
0.5, and at a bounding MCO pressure of 6.0 atm, the LPF_{MCO} is 0.83.

\*Nominal_{overall} M = MAR_{nominal} \times ARF_{nominal} \times RF_{nominal} \times LPF_{MCO}.

\*Bounding_{overall} M = (EF_{overall}) (nominal_{overall} M).

\*EF_{overall} = \exp \left( SNV_{95\%} \left( \sum_i \left[ \frac{\ln(EF_i)}{SNV_i} \right]^2 \right)^{1/2} \right)

ARF = airborne release fraction.
EF = error factor (i.e., bounding value divided by nominal value).
ln(EF) = natural log of EF.
LPF = leak path factor.
M = mass of material released.
MAR = material at risk.
MCO = multi-canister overpack.
RF = respirable fraction.
SNV = standard normal variable.
The bounding source term for the gaseous release is 1.2 g, which is the sum of $2.0 \times 10^4$ g from the aerodynamic entrainment of particulate and 1.2 g from vibration and shock.

### A3.4.2.1.3 Consequence Analysis

The dose calculation equation and data from Section 3.4.1 of the SNF Project FSAR are used to calculate the dose to the onsite receptor.

$$D_{\text{onsite}} = M \times \frac{\chi}{Q} \times BR \times UD \times LPF_{\text{building}}$$

$$= (1.2 \text{ g U})(1.14 \times 10^{-2} \text{ s/m}^3)(3.33 \times 10^{-4} \text{ m}^3/\text{s})(4.38 \times 10^5 \text{ rem/g U})(1.0)$$

$$= 1.9 \text{ rem (1.9 } \times 10^{-2} \text{ Sv)}.$$  

where

- $D_{\text{onsite}}$ = committed effective dose equivalent (rem)
- $M$ = mass of respirable airborne material released (g U)
- $\chi/Q$ = time-integrated atmospheric transport factor (s/m$^3$)
- $BR$ = breathing rate (m$^3$/s)
- $UD$ = dose per unit mass of uranium (rem/g U)
- $LPF_{\text{building}} = 1.0.$

The dose consequences at the remaining receptor sites are calculated in the same manner and are shown in Table A3-10.

### Table A3-10. Dose Calculation Summary for Mechanical Damage of a Multi-Canister Overpack.

<table>
<thead>
<tr>
<th>Receptor location (distance, direction)</th>
<th>Duration (hours)</th>
<th>Unmitigated dose$^a$ (rem (Sv))</th>
<th>Evaluation guideline$^b$/release limits (rem (Sv)) unlikely$^c$</th>
<th>Mitigated dose (rem (Sv))</th>
</tr>
</thead>
<tbody>
<tr>
<td>Onsite (with building effects) (100 m E)</td>
<td>&lt;1</td>
<td>1.9 (1.9 E-02)</td>
<td>10 (1.0 E-01)</td>
<td>--</td>
</tr>
<tr>
<td>Highway 240$^d$ (9,280 m W)</td>
<td>&lt;1</td>
<td>4.0 E-03 (4.0 E-05)</td>
<td>--</td>
<td>--</td>
</tr>
<tr>
<td>Hanford Site boundary (17,390 m E)</td>
<td>&lt;1</td>
<td>2.2 E-03 (2.2 E-05)</td>
<td>5 (5.0 E-02)</td>
<td>--</td>
</tr>
</tbody>
</table>

$^a$Fifty-year committed effective dose equivalent.

$^b$Evaluation guideline for onsite (100 m) receptor only.

$^c$Unmitigated frequency for these events is unlikely ($>10^{-4}$ to $\leq 10^{-2}$ per year) or less.

$^d$Provided for information only.
Because the consequences of the unmitigated accident do not exceed offsite release limits, no mitigated consequences were calculated for the offsite doses. No mitigated dose calculation was performed for the onsite receptor because the onsite consequences are within the evaluation guidelines for an unlikely event.

A3.4.2.1.4 Comparison to Guidelines. The unmitigated radiological offsite dose for all mechanical damage events is below offsite release limits while the unmitigated onsite dose is within evaluation guidelines for unlikely events.

A3.4.2.1.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls. This section identifies the safety SSCs and provides a brief description of their safety functions.

Drop of a Multi-Canister Overpack.

D1 Cask–MCO drop from receiving crane onto receiving area
- Safety-significant SSCs
  - MCO transportation cask — Provide structural protection for confinement capability (note that the MCO transportation cask has been designated safety class at other SNF Project facilities)
  - Receiving crane structure and hoist — Provides design features to protect assumptions for MCO drop frequencies; designation required, as well, by NRC equivalency important-to-safety Category B for SSCs that handle SNF.

D2 Cask–MCO drop from receiving crane into cask receiving pit
- Safety-significant SSCs
  - MCO transportation cask — Provide structural protection for confinement capability (note that the MCO transportation cask has been designated safety class at other SNF Project facilities)
  - Receiving crane structure and hoist — Provide design features to protect assumptions for MCO drop frequencies; designation required, as well, by NRC equivalency important-to-safety Category B for SSCs that handle SNF.

D3 Cask–MCO drop from receiving crane into FFTF pit or MHM maintenance pit
- Safety-significant SSCs
HNF-3553 REV 0
Annex A — Canister Storage Building

D4 Drop of MCO by MHM onto edge of cask receiving pit

- Safety-significant SSCs
  - MCO shell, locking ring, and shield plug — Provide structural protection for confinement capability (note that the MCO shell has been designated safety-class at other SNF Project facilities)
  - MHM structural components and MHM MCO hoist and grapple — Provide design features to protect assumptions for MCO drop frequencies; designation required, as well, by NRC equivalency important-to-safety Category B for SSCs that handle SNF.

D5 Drop of MCO by MHM back into cask

- Safety-significant SSCs
  - MCO shell, locking ring, and shield plug — Provide structural protection for confinement capability (note that the MCO shell has been designated safety-class at other SNF Project facilities)
  - MHM structural components and MHM MCO hoist and grapple — Provide design features to protect assumptions for MCO drop frequencies; designation required, as well, by NRC equivalency important-to-safety Category B for SSCs that handle SNF.

D6 Drop of MCO within MHM turret onto MHM turret deck

- Safety-significant SSCs
  - MCO shell, locking ring, and shield plug — Provide structural protection for confinement capability (note that the MCO shell has been designated safety-class at other SNF Project facilities)
  - MHM structural components and MHM MCO hoist and grapple — Provide design features to protect assumptions for MCO drop frequencies; designation...
required, as well, by NRC equivalency important-to-safety Category B for
SSCs that handle SNF.

D7 Drop of MCO from MHM onto CSB operating area floor or onto storage tube covers of
vault 2 or vault 3 of CSB

- Safety-significant SSCs
  - MHM structural components and MHM MCO hoist and grapple — Provide
design features to protect assumptions for MCO drop frequencies; designation
required, as well, by NRC equivalency important-to-safety Category B for
SSCs that handle SNF.

D8 Drop of MCO from MHM onto or into MHM maintenance pit or tube plug exchange
facility

- Safety-significant SSCs
  - MHM structural components and MHM MCO hoist and grapple — Provide
design features to protect assumptions for MCO drop frequencies; designation
required, as well, by NRC equivalency important-to-safety Category B for
SSCs that handle SNF.

D9 Drop of MCO from MHM onto or into FFTF pit

- Safety-significant SSCs
  - MHM structural components and MHM MCO hoist and grapple — Provide
design features to protect assumptions for MCO drop frequencies; designation
required, as well, by NRC equivalency important-to-safety Category B for
SSCs that handle SNF.

D10 Drop of MCO by MHM into storage tube or sampling/weld station

- Safety-significant SSCs
  - MCO shell, locking ring, and shield plug — Provide structural protection for
confinement capability (note that the MCO shell has been designated safety-
class at other SNF Project facilities)
  - MHM structural components and MHM MCO hoist and grapple — Provide
design features to protect assumptions for MCO drop frequencies; designation
required, as well, by NRC equivalency important-to-safety Category B for
SSCs that handle SNF.
Drop of MCO by MHM onto another MCO located in the bottom of a storage tube (no intermediate impact absorber involved)

- Safety-significant SSCs
  - MCO shell, locking ring, and shield plug — Provide structural protection for confinement capability (note that the MCO shell has been designated safety-class at other SNF Project facilities)
  - MHM structural components and MHM MCO hoist and grapple — Provide design features to protect assumptions for MCO drop frequencies; designation required, as well, by NRC equivalency important-to-safety Category B for SSCs that handle SNF.

Drop of MCO by MHM onto storage tube plug because of inadvertent turret rotation

- Safety-significant SSCs
  - MHM structural components and MHM MCO hoist and grapple — Provide design features to protect assumptions for MCO drop frequencies; designation required, as well, by NRC equivalency important-to-safety Category B for SSCs that handle SNF.

Drop of MCO by MHM onto edge of storage tube

- Safety-significant SSCs
  - MCO shell, locking ring, and shield plug — Provide structural protection for confinement capability (note that the MCO shell has been designated safety-class at other SNF Project facilities)
  - MHM structural components and MHM MCO hoist and grapple — Provide design features to protect assumptions for MCO drop frequencies; designation required, as well, by NRC equivalency important-to-safety Category B for SSCs that handle SNF.

Drop of MCO by MHM onto edge of sampling/weld station

- Safety-significant SSCs
  - MCO shell, locking ring, and shield plug — Provide structural protection for confinement capability (note that the MCO shell has been designated safety-class at other SNF Project facilities).
MHM structural components and MHM MCO hoist and grapple — Provide design features to protect assumptions for MCO drop frequencies; designation required, as well, by NRC equivalency important-to-safety Category B for SSCs that handle SNF.

**Shear of a Multi-Canister Overpack.**

S1 Shear of cask–MCO by moving receiving crane while cask–MCO is partially lowered into cask receiving pit, or shear of cask–MCO because of collision between MHM and receiving crane as receiving crane lowers cask–MCO into cask receiving pit

- Safety-significant SSCs
  - MCO transportation cask — Provide structural protection for confinement capability (note that the MCO transportation cask has been designated safety class at other SNF Project facilities).

S2 Shear of MCO by MHM turret rotation with MCO partially retrieved into MHM at cask receiving pit

- Safety-significant SSCs
  - MCO shell, locking ring, and shield plug — Provide structural protection for confinement capability (note that the MCO shell has been designated safety-class at other SNF Project facilities)

S3 Shear of MCO by MHM translational movement with MCO only partially retrieved into MHM at the cask receiving pit

- Safety-significant SSCs
  - MCO shell, locking ring, and shield plug — Provide structural protection for confinement capability (note that the MCO shell has been designated safety-class at other SNF Project facilities).

S4 Shear of MCO while MCO is partially retrieved from or inserted into cask receiving pit, storage tube, or sampling/weld station during a seismic event

No safety SSCs or TSRs are required to prevent or mitigate this accident.
S5 Shear of MCO by MHM translational movement with MCO only partially retrieved into MHM turret at storage tube or sampling/weld station

- Safety-significant SSCs
  - MCO shell, locking ring, and shield plug — Provide structural protection for confinement capability (note that the MCO shell has been designated safety-class at other SNF Project facilities).

Impacts to a Multi-Canister Overpack Other than Drops or Shears.

O1 Drop of cask lifting yoke onto the cask lid, or drop of the cask lid onto MCO

- Safety-significant SSCs
  - MCO transportation cask — Provide structural protection for confinement capability (note that the MCO transportation cask has been designated safety-class at other SNF Project facilities)
  - MCO shell, locking ring, and shield plug — Provide structural protection for confinement capability (note that the MCO shell has been designated safety-class at other SNF Project facilities)
  - Receiving crane structure and hoist — Provides design features to protect assumptions for MCO drop frequencies; designation required, as well, by NRC equivalency important-to-safety Category B for SSCs that handle SNF.

O2 Collapse of CSB structure and impact to MCO during seismic event

- Safety-class SSCs
  - Standard and overpack storage tubes, carbon steel basemat embeds, tube base assemblies, operating deck, and vault — Prevent collapse of CSB during seismic event

O3 Drop of cask receiving pit shield hatch assembly's center plate by MHM onto MCO, or drop of storage tube plug or sampling/weld station cover plug by MHM onto MCO in the storage tube or sampling/weld station pit

No safety SSCs or TSRs are required to prevent or mitigate this accident.
O4  MHM fall onto operating deck resulting in major structural damage to deck

- Safety-class SSCs
  - Standard and overpack storage tubes, carbon steel basemat embeds, tube base assemblies, operating deck, and vault — Prevent collapse of CSB during seismic events up to the DBE.
  - Rail frogs, rails, and MHM seismic restraints

- Safety-significant SSCs
  - Operating area shelter (waiver 1 to HNF-PRO-704)
  - MHM (structural components) (waiver 2 to HNF-PRO-704).

O5  Drop of intermediate impact absorber onto MCO

- Safety-significant SSCs
  - MCO shell, locking ring, and shield plug — Provide structural protection for criticality geometry control under normal operating conditions (note that the MCO shell has been designated safety-class at other SNF Project facilities)
  - MHM structural components and MHM MCO hoist and grapple — Provide design features to protect assumptions for MCO drop frequencies; designation required, as well, by NRC equivalency important-to-safety Category B for SSCs that handle SNF.

The SSCs, TSR controls, and defense-in-depth features designated to mitigate mechanical damage to an MCO from a drop event are summarized in Table A3-11. The SSCs, TSR controls, and defense-in-depth features designated to mitigate mechanical damage to an MCO from a shear event or other impact are summarized in Table A3-12. NRC important-to-safety category SSCs and defense-in-depth features also are included for each specific accident in Tables 3-11 and 3-12.
**Table A3-11. Summary of Safety Features Required to Prevent or Mitigate Mechanical Damage of the Multi-Canister Overpack from a Drop Accident.** (7 sheets)

<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designator*</th>
<th>General function</th>
<th>Safety feature and safety classification$^b$</th>
<th>NRC ITS category$^b$</th>
</tr>
</thead>
</table>
| 4 D1, D2, Cask-MCO drop from the receiving crane | TV-G-13, SA-G-13a, SA-G-13b | Reduce the likelihood of a drop and breach of an MCO | Safety-significant SSCs:  <ul>  
- Receiving crane structure and hoist  
- Transportation cask  
</ul> Safety-significant defense in depth:  <ul>  
- Cask lifting yoke  
- Cask receiving impact absorber (D2 only)  
</ul> Defense in depth:  <ul>  
- Crane height limiting device setpoints to limit lift height and prevent two-blocking  
- The operator of the receiving crane are trained and qualified to perform their duties safely, which includes following procedures for safe handling of the transportation cask.  
- Regular maintenance is performed on the transporter to ensure it is in good working order.  
- Maintenance and operations manuals and details are provided by crane vendors.  
- The hoist design includes interlocks to preclude lift and horizontal motion at same time, dual brakes, no free fall capacity.  
- MCO for confinement.  
</ul> | B |
| 9 D3. Drop of the cask-MCO from the receiving crane into the FFTF or MHM maintenance pit | SA-G-13c | Reduce the likelihood of a drop and breach of an MCO | Safety-significant SSCs:  <ul>  
- Transportation cask  
- Receiving crane structure and hoist  
</ul> Defense in depth:  <ul>  
- Interlock and fortress key for controlling receiving crane location.  
- Receiving crane positioning and interlock control system  
- Lifting devices used at the CSB are designed to handle the loads they will carry.  
- Personnel are trained to facility-specific procedures in the proper handling of the transportation cask, MCO, receiving crane, gantry, and MHM.  
- Regular maintenance is performed to ensure it is in good working order.  
- Maintenance and operators manuals and details are provided.  
</ul> | B |
Table A3-11. Summary of Safety Features Required to Prevent or Mitigate Mechanical Damage of the Multi-Canister Overpack from a Drop Accident. (7 sheets)

<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designator</th>
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<th>NRC ITS category</th>
</tr>
</thead>
<tbody>
<tr>
<td>D4, D5.</td>
<td>SA-G-13d</td>
<td>Reduce the likelihood of a drop and breach of an MCO</td>
<td>Safety-significant SSCs:</td>
<td>B</td>
</tr>
<tr>
<td>Drop of</td>
<td>SA-G-13e</td>
<td></td>
<td>• MHM structural components and MHM MCO hoist and grapple</td>
<td></td>
</tr>
<tr>
<td>MCO onto</td>
<td></td>
<td></td>
<td>• MCO shell, locking ring, and shield plug</td>
<td></td>
</tr>
<tr>
<td>the cask</td>
<td></td>
<td></td>
<td>Safety-significant defense-in-depth features:</td>
<td></td>
</tr>
<tr>
<td>receiving pit</td>
<td></td>
<td></td>
<td>• Shield hatch and MCO guide assembly on top of the cask receiving pit</td>
<td></td>
</tr>
<tr>
<td>or back into</td>
<td></td>
<td></td>
<td>• MCO centering guide installed in the MHM</td>
<td></td>
</tr>
<tr>
<td>the cask by</td>
<td></td>
<td></td>
<td>• Cask receiving impact absorber</td>
<td></td>
</tr>
<tr>
<td>the MHM</td>
<td></td>
<td></td>
<td>Defense in depth:</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• MHM interlocks (P57, P61, P62, P63, P65, and P66)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Personnel are trained to facility-specific procedures in the proper handling of the transportation cask, MCO, receiving crane, gantry, and MHM.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Regular maintenance is performed to ensure it is in good working order.</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>• Maintenance and operators manuals and details are provided.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• MHM ventilation.</td>
<td></td>
</tr>
<tr>
<td>D6. Drop of</td>
<td>OA-G-13a</td>
<td>Reduce the likelihood of a drop and breach of an MCO</td>
<td>Safety-significant SSCs:</td>
<td>B</td>
</tr>
<tr>
<td>MCO within</td>
<td></td>
<td></td>
<td>• MHM structural components and MHM MCO hoist and grapple</td>
<td></td>
</tr>
<tr>
<td>MIIM</td>
<td></td>
<td></td>
<td>• MCO shell, locking ring, and shield plug</td>
<td></td>
</tr>
<tr>
<td>cask-MCO</td>
<td></td>
<td></td>
<td>Safety-significant defense-in-depth features:</td>
<td></td>
</tr>
<tr>
<td>tube onto</td>
<td></td>
<td></td>
<td>• The MHM grapple is designed with a mechanical lock such that it should not be able to open while a load is suspended from it</td>
<td></td>
</tr>
<tr>
<td>MIIM turret deck</td>
<td></td>
<td></td>
<td>• MHM interlocks (P57, P61, P62, P63, P65, and P66)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Lifting devices used at the CSB are designed to handle the loads they will carry.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Personnel are trained to facility-specific procedures in the proper handling of the transportation cask, MCO, receiving crane, gantry, and MIIM.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Regular maintenance is performed to ensure it is in good working order.</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>• Maintenance and operators manuals and details are provided.</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>• MHM ventilation.</td>
<td></td>
</tr>
</tbody>
</table>
Table A3-11. Summary of Safety Features Required to Prevent or Mitigate Mechanical Damage of the Multi-Canister Overpack from a Drop Accident. (7 sheets)

<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designator</th>
<th>General function</th>
<th>Safety feature and safety classification</th>
<th>NRC ITS category</th>
</tr>
</thead>
</table>
| D7. Drop of the MCO from the MHM onto CSB operating floor or storage covers for CSB vaults 2 or 3 | OA-G-13b | Prevent the MCO from dropping at these specific locations | Safety-significant SSCa:  
- MHM structural components and MHM MCO hoist and grapple  
Defense in depth:  
- The MHM grapple is designed with a mechanical lock such that it should not be able to open while a load is suspended from it.  
- MHM interlocks (P57, P61, P62, P63, P65, P66)  
- Lifting devices used at the CSB are designed to handle the loads they will carry.  
- Personnel are trained to facility-specific procedures in the proper handling of the transportation cask, MCO, receiving crane, gantry, and MHM.  
- Regular maintenance is performed to ensure it is in good working order.  
- Maintenance and operators manuals and details are provided.  
- MHM ventilation. | B |
## Table A3-11. Summary of Safety Features Required to Prevent or Mitigate Mechanical Damage of the Multi-Canister Overpack from a Drop Accident. (7 sheets)

<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designator</th>
<th>General function</th>
<th>Safety feature and safety classification</th>
<th>NRC ITS category</th>
</tr>
</thead>
<tbody>
<tr>
<td>D8. Drop of the MCO from the MHM onto edge of or into the MHM maintenance pit or tube plug exchange facility</td>
<td>SA-G-13f</td>
<td>Prevent the MCO from dropping at these specific locations</td>
<td>Safety-significant SSCs:</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• MHM structural components and MHM MCO hoist and grapple</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>Defense in depth:</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• The MHM grapple is designed with a mechanical lock such that it should not be able to open while a load is suspended from it.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• MHM interlock (P2) to prevent MHM movement with tube plug in tube plug cavity.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• MHM interlocks (P57, P61, P62, P63, P65, and P66) and grapple design to protect against loss of MCO confinement function.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• MHM interlocks (P26, P85) to prevent turret rotation to MCO hoist position at tube plug exchange facility.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Lifting devices used at the CSB are designed to handle the loads they will carry.</td>
<td></td>
</tr>
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<td></td>
<td>• Personnel are trained to facility-specific procedures in the proper handling of the transportation cask, MCO, receiving crane, gantry, and MHM.</td>
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<td></td>
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<td></td>
<td>• Regular maintenance is performed to ensure it is in good working order.</td>
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<td></td>
<td></td>
<td></td>
<td>• Maintenance and operation manuals and details are provided.</td>
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<td></td>
<td>• MHM ventilation.</td>
<td></td>
</tr>
<tr>
<td>D9. Drop of the MCO from the MHM onto edge of or into FFTF maintenance pit</td>
<td>SA-G-13g</td>
<td>Prevent the MCO from dropping at these specific locations</td>
<td>Safety-significant SSCs:</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• MHM structural components and MHM MCO hoist and grapple</td>
<td></td>
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<td></td>
<td>Defense in depth:</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>• The MHM grapple is designed with a mechanical lock such that it should not be able to open while a load is suspended from it.</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>• MHM interlock (P2) to prevent MHM movement with tube plug in tube plug cavity.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• MHM interlocks (P57, P61, P62, P63, P65, and P66) and grapple design to protect against loss of MCO confinement function.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Lifting devices used at the CSB are designed to handle the loads they will carry.</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>• Personnel are trained to facility-specific procedures in the proper handling of the transportation cask, MCO, receiving crane, gantry, and MHM.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Regular maintenance is performed to ensure it is in good working order.</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>• Maintenance and operators manuals and details are provided.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• MHM ventilation.</td>
<td></td>
</tr>
</tbody>
</table>

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March 2000
### Table A3-11: Summary of Safety Features Required to Prevent or Mitigate Mechanical Damage of the Multi-Canister Overpack from a Drop Accident.

<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designator</th>
<th>General function</th>
<th>Safety feature and safety classification</th>
<th>NRC ITS category</th>
</tr>
</thead>
<tbody>
<tr>
<td>D10, D11, D13, Drop of the MCO onto or into the storage tube or the overpack storage tube or into the sampling/weld station</td>
<td>OA-G-13c</td>
<td>Prevent the MCO from breaching</td>
<td>Safety-significant SSCs:</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td>OA-G-13d</td>
<td></td>
<td>- MHM structural components and MHM MCO hoist and grapple</td>
<td></td>
</tr>
<tr>
<td></td>
<td>WS-G-13a</td>
<td></td>
<td>- MCO shell, locking ring, and shield plug</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td><strong>Safety-significant defense-in-depth features:</strong></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- Storage tube bottom impact absorber</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>- Storage tube intermediate impact absorber</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>- Sampling/weld station impact absorber</td>
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<tr>
<td></td>
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<td></td>
<td>- The lower flange of the storage tube, which has a 45° incline</td>
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<tr>
<td></td>
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<td></td>
<td>- The shield halves with a 75° incline</td>
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<tr>
<td></td>
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<td></td>
<td>- MCO centering guide installed in MHM</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>- Interface guide ring funnel is in place before arrival of the MCO in the MHM over a storage tube.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td><strong>Defense in depth:</strong></td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- The MHM grapple is designed with a mechanical lock such that it should not be able to open while a load is suspended from it.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- MHM interlocks (P57, P61, P62, P63, P65, and P66)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- Lifting devices used at the CSB are designed to handle the loads they will carry.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- Personnel are trained to facility-specific procedures in the proper handling of the transportation cask, MCO, receiving crane, gantry, and MHM.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- Regular maintenance is performed to ensure it is in good working order.</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>- Maintenance and operators manuals and details are provided.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- The hoist design includes interlocks to preclude lift and horizontal motion at same time, dual brakes, no free fall capacity.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- MHM ventilation system.</td>
<td></td>
</tr>
</tbody>
</table>
### Table A3-11. Summary of Safety Features Required to Prevent or Mitigate Mechanical Damage of the Multi-Canister Overpack from a Drop Accident. (7 sheets)

<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designatora</th>
<th>General function</th>
<th>Safety feature and safety classificationb</th>
<th>NRC ITS categoryb</th>
</tr>
</thead>
</table>
| D12. Drop of the MCO onto storage tube plug because of inadvertent turret rotation | OA-G-13b | Prevent the MCO from dropping onto this location | Safety-significant SSCs:  
  - MHM structural components and MHM MCO hoist and grapple  
  Safety-significant defense-in-depth features:  
  - Interface guide ring funnel is in place before arrival of the MCO in the MHM over a storage tube. | B |
|          |                       |                  | Defense in depth:  
  - The MHM grapple is designed with a mechanical lock such that it should not be able to open while a load is suspended from it.  
  - MHM interlocks (P57, P61, P62, P63, P65, and P66)  
  - Lifting devices used at the CSB are designed to handle the loads they will carry.  
  - Personnel are trained to facility-specific procedures in the proper handling of the transportation cask, MCO, receiving crane, gantry, and MHM.  
  - Regular maintenance is performed to ensure it is in good working order.  
  - Maintenance and operators manuals and details are provided.  
  - The hoist design includes interlocks to preclude lift and horizontal motion at same time, dual brakes, no free fall capacity  
  - MHM ventilation system. |
Table A3-11. Summary of Safety Features Required to Prevent or Mitigate Mechanical Damage of the Multi-Canister Overpack from a Drop Accident. (7 sheets)

<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designator*</th>
<th>General function</th>
<th>Safety feature and safety classificationb</th>
<th>NRC ITS categoryb</th>
</tr>
</thead>
</table>
| D14. Drop of the MCO onto the edge of sampling/ weld station pit | WS-G-13b | Prevent the MCO from breaching | Safety-significant SSCs:  
- MCO shell, locking ring, and shield plug  
- MHM structural components and MHM MCO hoist and grapple  

Safety-significant defense-in-depth features:  
- Shield halves with a 75° incline  
- MCO centering guide installed in the MHM  

Defense in depth:  
- The MHM grapple is designed with a mechanical lock such that it should not be able to open while a load is suspended from it.  
- Interface guide ring funnel  
- MHM interlocks (P57, P61, P62, P63, P65, and P66)  
- Lifting devices used at the CSB are designed to handle the loads they will carry.  
- Personnel are trained to facility-specific procedures in the proper handling of the transportation cask, MCO, receiving crane, gantry, and MHM.  
- Regular maintenance is performed to ensure it is in good working order.  
- Maintenance and operators manuals and details are provided.  
- The hoist design includes interlocks to preclude lift and horizontal motion at same time, dual brakes, no free fall capacity.  
- MHM ventilation system. |

Note: Defense-in-depth features and safety-significant defense-in-depth features are not designated as NRC ITS.


bSSCs are classified per their function in mitigating or preventing specific accidents. SSCs may have other classifications based on their functions in other events.

CSB = Canister Storage Building.
FFTF = Fast Flux Test Facility.
ITS = important to safety.
MCO = multi-canister overpack.
MHM = multi-canister overpack handling machine.
NA = not applicable.
NRC = U.S. Nuclear Regulatory Commission.
SSC = structure, system, and component.
TSR = technical safety requirement.
Table A3-12. Summary of Safety Features Required to Prevent or Mitigate Mechanical Damage of the Multi-Canister Overpack from a Shear or Impact Accident. (6 sheets)

<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designator</th>
<th>General function</th>
<th>Safety feature and safety classification</th>
<th>NRC ITS category</th>
</tr>
</thead>
<tbody>
<tr>
<td>S1. Shear of cask-MCO by moving receiving crane or collision between the MHM and cask-MCO</td>
<td>SA-F-05, SA-F-07b, SA-F-07c</td>
<td>Provide passive structural protection for the MCO</td>
<td>Safety-significant SSCs: • Transportation cask</td>
<td>B</td>
</tr>
<tr>
<td></td>
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<td></td>
<td>Defense in depth: • Interlock P5 to inhibit receiving crane and MHM interaction</td>
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<td></td>
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<td></td>
<td>• Personnel are trained to procedures detailing the safe sequence of operations; these procedures prohibit interferences between the receiving crane and the MHM</td>
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<tr>
<td></td>
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<td>• The frog on the receiving crane's track has a stop</td>
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<td></td>
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<td></td>
<td>• The receiving crane has an auditory indication of its movement (i.e., alarm)</td>
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<td></td>
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<td></td>
<td>• The receiving crane is limited to relatively slow movement</td>
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<td></td>
<td></td>
<td>• The receiving crane design includes such features as restricted speed, load float, micropositioner, and auditory movement alarms</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>• Regular maintenance is performed on the receiving crane to ensure it is in good working order</td>
<td></td>
</tr>
<tr>
<td>S2. Shear of MCO by rotation</td>
<td>SA-E-07, OA-E-07, WS-E-07</td>
<td>Prevent MCO from breaching</td>
<td>Safety-significant SSCs: • MCO shell, locking ring, and shield plug</td>
<td>B</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Defense in depth: • MHM interlock (P9), sensors, and switches</td>
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</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• MHM interlock (P6, P80), sensors, and switches</td>
<td></td>
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<tr>
<td></td>
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<td></td>
<td>• MHM seismic restraints</td>
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<td></td>
<td>• Seismic detection and MHM power-disconnect system</td>
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<td></td>
<td>• Personnel are trained in sitewide and facility-specific emergency response procedures.</td>
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<td></td>
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<td></td>
<td>• The MHM provides active, filtered ventilation at its open interface with both the cask receiving pit and the sampling/weld station.</td>
<td></td>
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<tr>
<td></td>
<td></td>
<td></td>
<td>• The MHM is designed to ASME NOG-1 to preclude tipping.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• The MHM has an auditory indication of its movement (i.e., alarms).</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• The MHM is limited to relatively slow movement</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• The MHM is provided with a backup grapple disengagement capability.</td>
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<tr>
<td></td>
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<td></td>
<td>• Personnel are trained to procedures detailing the safe sequence of operations; these procedures prohibit interferences between the receiving crane and the MHM</td>
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<tr>
<td></td>
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<td></td>
<td>• MHM ventilation system.</td>
<td></td>
</tr>
</tbody>
</table>
Table A3-12. Summary of Safety Features Required to Prevent or Mitigate Mechanical Damage of the Multi-Canister Overpack from a Shear or Impact Accident. (6 sheets)

<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designator*</th>
<th>General function</th>
<th>Safety feature and safety classificationb</th>
<th>NRC ITS categoryb</th>
</tr>
</thead>
</table>
| S3. Shear MCO by moving MHM with MCO partially retrieved from cask | SA-F-07d | Prevent MCO translational shear | Safety-significant SSCs:  
- MCO shell, locking ring, and shield plug  
Safety-significant defense-in-depth features:  
- MHM interlock (P2) to prevent MHM movement with tube plug in tube plug cavity.  
Defense in depth:  
- MHM interlocks (P3, P6, P8, P80), sensors, and switches  
- Personnel are trained in sitewide and facility-specific emergency response procedures  
- The MHM provides active, filtered ventilation at its open interface with both the cask receiving pit and the sampling/weld station  
- The MHM is designed to ASME NOG-1’ to preclude tipping  
- The MHM has an auditory indication of its movement (i.e., alarms)  
- The MHM is provided with a backup grapple disengagement capability  
- Personnel are trained to procedures detailing the safe sequence of operations; these procedures prohibit interferences between the receiving crane and the MHM. | B |
Table A3-12. Summary of Safety Features Required to Prevent or Mitigate Mechanical Damage of the Multi-Canister Overpack from a Shear or Impact Accident. (6 sheets)

<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designator&lt;sup&gt;a&lt;/sup&gt;</th>
<th>General function</th>
<th>Safety feature and safety classification&lt;sup&gt;b&lt;/sup&gt;</th>
<th>NRC ITS category&lt;sup&gt;b&lt;/sup&gt;</th>
</tr>
</thead>
</table>
| S4. Shear MCO by MHM during a seismic event | OA-F-07b, WS-F-07b, SA-F-07e | Prevent MCO translational shear | Safety-significant defense-in-depth features:  
- MHM seismic restraints (trolley, turret, and bridge)  
- MHM rails and rail frogs  
- MHM interlock (P6, P21), sensors, and switches  
Defense in depth:  
- MHM interlocks (P3, P6, P8, P26, P80), sensors, and switches  
- Seismic detection and MHM power-disconnect system  
- Personnel are trained in sitewide and facility-specific emergency response procedures  
- The MHM provides active, filtered ventilation at its open interface with both the cask receiving pit and the sampling/weld station  
- The MHM is designed to ASME NOG-1<sup>c</sup> to preclude tipping  
- The MHM has an auditory indication of its movement (i.e., alarms)  
- The MHM is provided with a backup grapple disengagement capability  
- Personnel are trained to procedures detailing the safe sequence of operations; these procedures prohibit interferences between the receiving crane and the MHM. | |

<sup>a</sup> OA-F-07b, WS-F-07b, SA-F-07e

<sup>b</sup> NRC ITS Category

<sup>c</sup> ASME NOG-1
### Table A3-12. Summary of Safety Features Required to Prevent or Mitigate Mechanical Damage of the Multi-Canister Overpack from a Shear or Impact Accident. (6 sheets)

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</tr>
</thead>
<tbody>
<tr>
<td>S5. Shear MCO by moving MHM at storage tube or sampling/weld station</td>
<td>OA-F-07a, WS-F-07a</td>
<td>Prevent MCO translational shear</td>
<td><strong>Safety-significant SSCs:</strong>&lt;br&gt;- MCO shell, locking ring, and shield plug&lt;br&gt;<strong>Safety-significant defense-in-depth features:</strong>&lt;br&gt;- MHM interlock (P2), sensors, and switches&lt;br&gt;<strong>Defense in depth:</strong>&lt;br&gt;- MHM interlocks (P3, P6, P8, P80), sensors, and switches&lt;br&gt;- Personnel trained in sitewide and facility-specific emergency response procedures&lt;br&gt;- MHM provides active, filtered ventilation at its open interface with both the cask receiving pit and the sampling/weld station&lt;br&gt;- MHM is designed to ASME NOG-I to preclude tipping&lt;br&gt;- MHM has an auditory indication of its movement (i.e., alarms)&lt;br&gt;- MHM provided with a backup grapple disengagement capability&lt;br&gt;- Personnel trained to procedures detailing the safe sequence of operations; these procedures prohibit interferences between the receiving crane and the MHM&lt;br&gt;- MHM ventilation system.</td>
<td>B</td>
</tr>
</tbody>
</table>

| O1. Drop yoke onto cask lid or drop cask lid onto MCO | TV-G-03a, SA-G-03a, SA-G-03c | Prevent MCO from damage | **Safety-significant SSCs:**<br>- Transportation cask<br>- MCO shell, locking ring, and shield plug<br>- Receiving cask structure and hoist<br>**Defense in depth:**<br>- The operators of the receiving crane are trained and qualified to perform their duties safely, which includes following procedures for safe handling of the transportation cask<br>- Regular maintenance is performed on the transporter to ensure it is in good working order<br>- Maintenance and operations manuals and details are provided by crane vendors<br>- The hoist design includes<br>- Interlocks to preclude lift and horizontal motion at same time<br>- Dual brakes<br>- No free fall capacity | B |
Table A3-12. Summary of Safety Features Required to Prevent or Mitigate Mechanical Damage of the Multi-Canister Overpack from a Shear or Impact Accident. (6 sheets)

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<thead>
<tr>
<th>Accident</th>
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<th>NRC ITS category</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. O2, O4. Drop of CSB structural component onto an MCO or other safety-class structure</td>
<td>TV-G-03b, SA-G-03b, OA-G-03a, OA-G-03b, WS-G-03a, WS-G-03b</td>
<td>Prevent collapse of CSB structures or loss of structural integrity of the facility during NPH events</td>
<td>Safety-class SSCs:</td>
<td>A</td>
</tr>
<tr>
<td>2. O2, O4. Drop of CSB structural component onto an MCO or other safety-class structure</td>
<td></td>
<td></td>
<td>- Standard and overpack storage tubes, carbon steel basement embeds, tube base assemblies, operating deck, and vault - Rail frogs, rails, MHM seismic restraints (O4 only)</td>
<td>A</td>
</tr>
<tr>
<td>3. OR MHM fall onto the operating deck</td>
<td></td>
<td></td>
<td>Safety-significant SSCs:</td>
<td>B</td>
</tr>
<tr>
<td>4. OR MHM fall onto the operating deck</td>
<td></td>
<td></td>
<td>- Operating area shelter (waiver 1 to HNF-PRO-704*) - MHM (structural) (waiver 2 to HNF-PRO-704*) - Support area building foundation</td>
<td>B</td>
</tr>
<tr>
<td>5. OR MHM fall onto the operating deck</td>
<td></td>
<td></td>
<td>Defense in depth:</td>
<td>NA</td>
</tr>
<tr>
<td>6. OR MHM fall onto the operating deck</td>
<td></td>
<td></td>
<td>- Seismic detection and MHM power-disconnect to limit MHM travel - Sampling/weld station MCO support structure - Personnel are trained in sitewide and facility-specific emergency response procedures - The facility provides shelter for workers</td>
<td></td>
</tr>
<tr>
<td>7. O3. Drop by MHM of cask receiving pit shield hatch assembly, storage tube plug, or sampling/weld station cover plate</td>
<td>OA-G-14b, SA-G-03d</td>
<td>Designed to prevent damage to MCO when dropped</td>
<td>Defense in depth:</td>
<td></td>
</tr>
<tr>
<td>8. O3. Drop by MHM of cask receiving pit shield hatch assembly, storage tube plug, or sampling/weld station cover plate</td>
<td></td>
<td></td>
<td>- The MHM grapple is designed with a mechanical lock such that it should not be able to open while a load is suspended from it - MHM grapple is designed to handle the loads they will carry - Personnel are trained to facility-specific procedures in the proper handing of the MCO and MHM - Qualified operators</td>
<td></td>
</tr>
</tbody>
</table>
### Table A3-12. Summary of Safety Features Required to Prevent or Mitigate Mechanical Damage of the Multi-Canister Overpack from a Shear or Impact Accident. (6 sheets)

<table>
<thead>
<tr>
<th>Accident Description</th>
<th>Checklist Designator</th>
<th>General Function</th>
<th>Safety Feature and Safety Classification</th>
<th>NRC ITS Category</th>
</tr>
</thead>
<tbody>
<tr>
<td>Drop intermediate impact absorber on MCO</td>
<td>OA-G-14a</td>
<td>Limit impact forces on the MCO</td>
<td>Safety-significant SSCs: MCO shell, locking ring, and shield plug, MHM structural components and MHM MCO hoist and grapple</td>
<td>B</td>
</tr>
</tbody>
</table>

#### Defense in Depth:
- The MHM grapple is designed with a mechanical lock such that it should not be able to open while a load is suspended from it.
- MHM interlocks (P57, P61, P62, P63, P65, and P66)
- MHM grapple is designed to handle the loads they will carry.
- Personnel are trained in facility-specific procedures in the proper handling of the MCO and MHM.
- Qualified operators.
- Regular maintenance of the MHM is performed to ensure it is in good working order.
- The hoist design includes:
  - Interlocks to preclude lift and horizontal motion at the same time.
  - Dual brakes.
  - No free fall capacity.

Note: Defense-in-depth features and safety-significant defense-in-depth features are not designated as NRC ITS.

2. SSCs are classified per their function in mitigating or preventing specific accidents. SSCs may have other classifications based on their functions in other events.

CSB = Canister Storage Building.
ITS = Important to safety.
MCO = Multi-canister overpack.
MHM = Multi-canister overpack handling machine.
NPH = Natural phenomena hazard.
NRC = U.S. Nuclear Regulatory Commission.
SSC = Structure, system, and component.
TSR = Technical safety requirement.
A3.4.2.2 Gaseous Release from the Multi-Canister Overpack. This accident evaluates events that may result in releases of radioactive materials to the CSB facility caused by loss of the MCO confinement pressure boundary. The potential for hydrogen deflagrations caused by loss of confinement is discussed in Section A3.4.2.4. MCOs are backfilled with helium to a prescribed pressure just before being sealed for shipment from the CVDF to the CSB. The integrity of the mechanical seal also is checked and verified at the CVDF before the MCO is shipped to the CSB. The uncontrolled release of the MCO internal gas pressure, referred to as a pressurized gaseous release accident, results from failure of the MCO pressure boundary to confine the MCO gases at the CSB. A pressurized gaseous release would lead to the entrainment and release of fuel particulate from the MCO and the creation of a radiological hazard. The gases released from the MCO in this event are those in the MCO after it is processed at CVDF and those generated in the MCO at the CSB. These gases include helium, hydrogen, water vapor, and possibly some small concentration of oxygen.

Based on the results of the hazards analysis (HNF-SD-SNF-HIE-001) and shown in Table A3-1, the potential for gaseous releases from an MCO can be the result of operations at the CSB, chemical properties of the SNF, and a breach of confinement caused by drops or impacts. The bounding accidents selected for further evaluation include (1) overpressurization caused by radiolytic decomposition of water and aluminum hydroxide, and the generation of hydrogen from the chemical reaction of water with fuel inside the MCO; (2) sampling system equipment failures or operator error during sampling and backfilling operations at the CSB; and (3) breach of MCO confinement caused by drops or impacts of objects onto the MCO at the sampling/weld station. The potential for a drop of the MCO or cask-MCO and MCO or cask-MCO impacts has been addressed in Section A3.4.2.1. The scenarios considered in this accident analysis are assumed to not result in a deflagration of the gases within the MCO. Therefore, the bounding accident scenarios addressed in this section include overpressurization due to radiolytic decomposition and gaseous releases due to sampling system equipment failures or operator error during sampling and backfilling operations at the CSB. Internal and external deflagrations are addressed in subsequent DBA sections.

As discussed in the HNF-SD-SNF-TI-059, the radiological guidelines associated with the release of SNF are more limiting than the toxicological guidelines, thus, the toxicological consequences associated with the gases released in this event do not require mitigating features beyond those required by the radiological consequences. Additional information about the MCO gaseous release accidents can be found in SNF-3328.

A3.4.2.2.1 Scenario Development. The following describes the scenario development for each of the cases evaluated.

Overpressurization Due to Radiolytic Decomposition. MCOs arrive at the CSB from the CVDF in a mechanically sealed configuration with no pressure relief. Following operations at the CVDF the MCO is transported to the CSB. During the period encompassing closure at CVDF and awaiting transport to the CSB, transport to the CSB, and receipt at the CSB awaiting closure, there is the potential for the MCO to pressurize from radiolytic decomposition. This scenario
assumes that the MCO overpressurizes (i.e., pressure caused by radiolytic decomposition is sufficient to fail the mechanical seal) and releases gases and particulate to the CSB facility.

As discussed in Chapter A2.0, the mechanical seal is designed to withstand pressures up to 150 lb/in² (18.3 atm). The initial pressure within an MCO is 22 lb/in² absolute (1.5 atm). According to HNF-SD-SNF-TI-040, MCO Internal Gas Composition and Pressure during Interim Storage, internal MCO pressures following 40 years of storage could be as high as 76 lb/in² absolute (5.2 atm), which is well below the design pressure of the MCO. Therefore, insufficient pressures will be generated within the MCO to fail the mechanical seal (receipt conditions or conditions during sampling operations) or the welded seal (long-term or 40-year storage condition). Thus it can be concluded that insufficient pressures will be generated within an MCO from radiolytic decomposition to overpressurize an MCO and result in a release upon receipt at the CSB.

Gaseous Release During Sampling Operations. A gaseous release could occur at the CSB because of operator error during sampling and backfilling operations. This scenario assumes the incorrect installation or removal of the sampling and backfill equipment or a failure of the equipment resulting in a direct path for leakage into the CSB facility. Dropping of an object onto the sampling/weld station components is another potential initiator for a leak from the sampling system. A drop could shear the sampling system piping or otherwise fail the confinement capability of the sampling system pressure boundary. A confinement failure could also result from the sampling/weld station gantry crane, the MHM, or some other equipment moving horizontally into the sampling/weld station components and shearing the piping.

The sampling process begins when the MHM lowers the selected MCO into the sampling/weld station pit. The surface temperature of the MCO is monitored. If the surface of the top of the MCO is not sufficiently cool, the MCO is cooled using the sampling/weld station cooling cap to reduce the exposed portions of the MCO to a temperature consistent with standard industrial safety regulations. After removing temporary radiation shields and guard rails, the sampling hood is installed on the MCO to confine possible airborne contamination generated by an accidental release during sampling. The MCO sampling cart is connected to the local distributed control system and inert gas system. Piping on the sample cart connects to the sample gas accumulator through the sampling hood high-efficiency particulate air (HEPA) filter outlet using a quick-disconnect attachment. The flexible hose of the sample cart vent is connected to the sampling hood discharge, which dumps to the exhauster. This exhaust system fan establishes negative pressure within the hood relative to the operating area pressure to maintain air contamination control and air flow around the top of the MCO.

For MCO sampling, an MCO process valve operator is installed on the short tube port (port 2) of the MCO after the cover plate is removed. The process valve operator is connected with a quick-disconnect coupling to a flexible hose that connects to the inlet of the sample line HEPA filter. The remainder of the sample system piping is the flexible piping that connects the outlet of the sample line HEPA filter to the sample cart and the rigid piping in the sample cart that goes to the sample valve and the sample accumulator. This piping is used to draw a sample of gas
from the MCO and to pressurize the MCO using helium from the 120 lb/in² gauge CSB helium supply system. The sampling system piping between the refill valve and the MCO is pressurized during the MCO refill operation. The entire sampling process is a manual operation; the technician taking the sample is present at all times during sampling.

After completion of the sampling activities, the MCO is backfilled with helium if necessary to maintain internal pressure. The same equipment and process lines used during sampling are used to backfill the MCO. However, the sample valve is closed and the helium backfill valve is opened (see Figure A2-42).

The following scenario involves a failure of this process line while it is connected to the MCO with the MCO process valve open, leading to a blowdown of the gases inside the MCO. With only one process line connected to the MCO, no flow path through the MCO can exist to continuously remove particulate from the MCO for an extended period of time. To get a release, the break in a section of process piping must be upstream of the sample line HEPA filter. A release from a piping failure downstream of the sample line HEPA filter requires failure of the sample system HEPA filter as well. A release also is possible if the MCO valve operator is inadvertently removed before the process port is closed.

A leak within the confines of the sample hood would be quickly exhausted through the sampling/weld station HEPA filter to the CSB exhaust system plenum and out to the environment through the CSB stack. Leaks from the sample system outside of the sample hood would be to the CSB operating air.

The effective leak flow area is bounded by the flow area of the path through the MCO shield plug (1.0-in.-diameter). All of the sampling system piping outside of the sampling hood is less than 0.75-in. inside diameter except the sample line to the cart, which is 1 in. in diameter. The flexible hose inside the sample hood and the associated fittings have inside diameters less than 1.0 in. If the valve operator body is improperly secured to the top of the MCO, then the leak path flow area around the valve could be large enough to make the flow area of the shield plug the limiting flow area. Therefore, the largest leak path flow area possible would be for a leak into the sample hood. Sampling system leaks outside of the sample hood would have smaller flow area paths and would have traveled through the sample line HEPA filter.

Gaseous Release from Overpressurization Due to Equipment Failure. In addition to the gaseous release accident at the sampling/weld station, the overpressurization of the MCO by the inert gas system during reinerting of a monitored MCO after sampling also is considered. This accident is initiated by failure of the pressure regulator on the helium supply system while an MCO is in the sampling/weld station undergoing the MCO helium backfill operation. This scenario assumes the helium supply pressure is sufficient to breach the MCO shield plug seal. As with the sampling process, the entire backfilling process is a manual operation and the technician is present at all times.
A3.4.2.2 Source Term Analysis. For the gaseous release accident, the MAR (particulate available for release) depends on the amount of particulate generated between the time the fuel is washed at the K Basins and the time of the accident. The MCO gaseous release accident could only occur during the time the MCO sampling program is being conducted. The Technical Databook (HNF-SD-SNF-TI-015) states that the safety basis or bounding value for the mass of particulate in the MCO is 34.0 kg of uranium dioxide, which contain 30.0 kg of uranium. Because of the conservative methodology used to calculate the bounding value, 30 kg is considered representative of the 99th percentile. The Technical Databook (HNF-SD-SNF-TI-015) also states that the design or nominal value is 2.1 kg of uranium dioxide, which contain 1.85 kg of uranium.

The quantity of particulate actually released during a gaseous release accident depends on the initial aerosol concentration and subsequent particulate entrainment. The pressurized powder venting source term is not applicable, and a source term based upon aerodynamic entrainment, as described in Section A3.4.2.1.2, is used.

Based on the data reported in DOE-HDBK-3010-94, a bounding (95th percentile) ARR of $4 \times 10^{-5}$/h and an RF of 1.0 were selected. The contents of the MCO are intact fuel elements tightly packed in fuel baskets and pieces of fuel elements housed in scrap baskets. Particulate matter swept upward by streams of flowing gas within the MCO must take a tortuous path through the MCO and through the MCO shield plug to exit the MCO. For shielded powder, where the aerodynamic stresses are reduced by debris or exposure to static conditions, DOE-HDBK-3010-94 recommends an ARR of $4 \times 10^{-6}$/h and an RF of 0.2. These values were selected as nominal (50th percentile).

The pressurized gaseous release from the MCO occurs close to the MCO through a leak path with a flow area greater than or equal to the flow area through the MCO shield plug. Such a leak could be caused by complete severance of the pipe or flexible tubing within the sampling/weld station sample hood or by failure to tighten the hold down bolts on the MCO port valve operator. The duration of the blowdown depends on the flow area of the leak and the initial MCO pressure. The MCO depressurizes to atmospheric pressure in less than 1 minute, even for an MCO pressure of about 150 lb/in$^2$ gauge and a 0.25-in. hole. The MCO depressurizes in a much shorter time for lower pressures and larger leak paths. For the calculation of the source term, a nominal (50th percentile) value of 10 seconds and a bounding (95th percentile) time of 60 seconds were selected.

Using the methodology described in Section A3.4.1.3 and the bounding and nominal values identified above, the bounding (95th percentile) value for the material released through aerodynamic entrainment is calculated to be $2 \times 10^4$ g, as shown in Table A3-13.
Table A3-13. Aerodynamic Entrainment Source Term for the Gaseous Release from the Multi-Canister Overpack.

<table>
<thead>
<tr>
<th>Nominal</th>
<th>Bounding</th>
<th>EF</th>
<th>Percentile</th>
<th>SNV</th>
<th>In(EF)/SNV</th>
</tr>
</thead>
<tbody>
<tr>
<td>MAR</td>
<td>1.85 kg</td>
<td>30 kg</td>
<td>16.22</td>
<td>99</td>
<td>2.326</td>
</tr>
<tr>
<td>ARR</td>
<td>4.00 E-06/h</td>
<td>4.00 E-05/h</td>
<td>10.00</td>
<td>95</td>
<td>1.645</td>
</tr>
<tr>
<td>Time</td>
<td>10 s</td>
<td>60 s</td>
<td>6.00</td>
<td>95</td>
<td>1.645</td>
</tr>
<tr>
<td>RF</td>
<td>0.20</td>
<td>1.00</td>
<td>5.00</td>
<td>95</td>
<td>1.645</td>
</tr>
<tr>
<td>M</td>
<td>4.1 E-06 g (^{a})</td>
<td>2.0 E-04 g (^{b})</td>
<td>48(^{c})</td>
<td>95</td>
<td>1.645</td>
</tr>
</tbody>
</table>

Note: The 1.645 is the standard normal variable corresponding to the 95% upper confidence limit. For the 99% upper confidence limit, the corresponding value is 2.326.

\[^{a}\text{Nominal}_{\text{overall}}M = \text{MAR}_{\text{nominal}} \times \text{ARR}_{\text{nominal}} \times \text{RF}_{\text{nominal}} \times \text{time}_{\text{nominal}}\]

\[^{b}\text{Bounding}_{\text{overall}}M = (\text{EF}_{\text{overall}})(\text{nominal}_{\text{overall}}M)\]

\[^{c}\text{EF}_{\text{overall}} = \exp \left( \text{SNV}_{95\%} \left[ \sum_i \left[ \frac{\text{In}(\text{EF}_i)}{\text{SNV}_i} \right]^2 \right]^{\frac{1}{2}} \right)\]

ARR = airborne release rate.

EF = error factor (i.e., bounding value divided by nominal value).

ln(EF) = natural log of EF.

M = mass of material released.

MAR = material at risk.

RF = respirable fraction.

SNV = standard normal variable.

The aerosol concentration inside the MCO at the time of the release depends on prior MCO handling conditions. The powder in the MCO at rest could be ejected into the gas volume and suspended by the response of the underlying solid MCO and fuel substrate to vibration or jolting induced by impact or falling debris. According to DOE-HDBK-3010-94, the value of the ARF under such circumstances should exceed the value of the ARF for aerodynamic suspension alone but be less than the ARF for the free fall of powder. The powder undergoing vibration shock is bounced into the gas while subject to the same gas velocities as those for aerodynamic entrainment. Based on the discussion in DOE-HDBK-3010-94, a bounding (95th percentile) ARF of $1 \times 10^3$ and an RF of 1.0 were chosen for the suspension of powder-like surface contamination by shock vibration. A nominal (50th percentile) value of $4 \times 10^4$ was chosen for the ARF and a nominal value of 0.2 was chosen for the RF based on evaluation of the discussion in DOE-HDBK-3010-94. The pressure in the MCO at the start of the blowdown determines the fraction of the suspended particulate that is released. The range in initial pressure varies from 1.50 atm (corresponding to the helium fill pressure of 22 lb/in\(^2\) absolute at the CVDF) to a maximum pressure of 5.2 atm (as defined in the Technical Databook [HNF-SD-SNF-TI-015]).
To allow for some additional margin in the calculation, a pressure of 2.0 atm is considered
nominal (50th percentile) and corresponds to an LPF of 0.5, while a pressure of 6.0 atm is
considered bounding (95th percentile) and corresponds to an LPF of 0.83.

Unlike in a drop event, there is no clear mechanism identified in this accident to provide the
impact to suspend particulate within the MCO. However, normal handling of the MCO by the
MHM during transport from the storage tube to the sampling/weld station will produce some
vibration and jostling of the fuel and suspend some material. For the purposes of this analysis, it
is assumed that nominally 1% of the impact suspension concentration exists within the MCO at
the start of sampling. Therefore a value of 1% is assumed for the 50th percentile. In addition, a
value of 10% is assumed for the 95th percentile.

Using the methodology described in Section A3.4.1.3 and the bounding and nominal values
identified above, the bounding (95th percentile) value for the material released by impact is
calculated to be $2.7 \times 10^{-2}$ g, as shown in Table A3-14.

The bounding source term for the gaseous release is $2.7 \times 10^{-2}$ g, which is the sum of
$2.0 \times 10^{-4}$ g from the aerodynamic entrainment of particulate and $2.7 \times 10^{-2}$ g from vibration and
shock.

A3.4.2.2.3 Consequence Analysis. The radiological dose is calculated using the dose
calculation equation and data from Section 3.4.1 of the SNF Project FSAR.

The dose to the onsite receptor is calculated as follows:

$$D_{on site} = M \times \frac{X}{Q'} \times BR \times UD \times LPF_{building}$$

$$= (0.027 \text{ g U})(1.14 \times 10^{-2} \text{ s/m}^3)(3.33 \times 10^{-4} \text{ m}^3/\text{s})(4.38 \times 10^5 \text{ rem/g U})(1.0)$$

$$= 4.5 \times 10^{-2} \text{ rem (4.5 \times 10^{-4} Sv)}.$$
Table A3-14. Initial Suspended Material Source Term for the Gaseous Release from the Multi-Canister Overpack.

<table>
<thead>
<tr>
<th></th>
<th>Nominal</th>
<th>Bounding</th>
<th>EF</th>
<th>Percentile</th>
<th>SNV</th>
<th>In(EF)/SNV</th>
</tr>
</thead>
<tbody>
<tr>
<td>MAR</td>
<td>1.85 kg</td>
<td>30 kg</td>
<td>16.22</td>
<td>99</td>
<td>2.326</td>
<td>1.198</td>
</tr>
<tr>
<td>ARF</td>
<td>4.00 E-04</td>
<td>1.00 E-03</td>
<td>2.50</td>
<td>95</td>
<td>1.645</td>
<td>0.557</td>
</tr>
<tr>
<td>RF</td>
<td>0.20</td>
<td>1.00</td>
<td>5.00</td>
<td>95</td>
<td>1.645</td>
<td>0.978</td>
</tr>
<tr>
<td>Settling</td>
<td>1%</td>
<td>10%</td>
<td>10.00</td>
<td>95</td>
<td>1.645</td>
<td>1.400</td>
</tr>
<tr>
<td>LPF_{MCO}</td>
<td>0.50</td>
<td>0.83</td>
<td>1.67</td>
<td>95</td>
<td>1.645</td>
<td>0.311</td>
</tr>
</tbody>
</table>

Note: The 1.645 is the standard normal variable corresponding to the 95% upper confidence limit. For the 99% upper confidence limit, the corresponding value is 2.326.

\[
\text{LPF}_{MCO} = \frac{(\text{MCO pressure} - 1 \text{ atm})}{(\text{MCO pressure})}.
\]

At a nominal MCO pressure of 2.0 atm, the \( \text{LPF}_{MCO} \) is 0.5, and at a bounding MCO pressure of 6.0 atm, the \( \text{LPF}_{MCO} \) is 0.83.

\[\text{ARF}_{\text{overall}} = \exp\left(\text{SNV}_{95\%} \left( \sum_i \left[ \frac{\ln\left[E\left(E_{i}\right)\right]}{\text{SNV}_{i}} \right]^2 \right)^{1/2}\right)\]

ARF = airborne release fraction.
EF = error factor (i.e., bounding value divided by nominal value).
\( \ln(EF) \) = natural log of EF.
LPF = leak path factor.
M = mass of material released.
MAR = material at risk.
MCO = multi-canister overpack.
RF = respirable fraction.
SNV = standard normal variable.

The actual duration of the release is expected to be less than 1 hour for all receptors, therefore the applied \( \chi/Q \) value is conservatively chosen for a 1-hour duration. The estimated bounding doses to receptors at several locations have been calculated and the results summarized in Table A3-15.
### A3.4.2.2.4 Comparison to Guidelines

The annual frequency of a gaseous release from the sampling system is estimated to be the sum of the frequency of a leak from the MCO sampling system pressure boundary configuration that exists during sampling. This includes (1) a leak resulting from failure to properly seal the MCO valve operator at the sampling/weld station, (2) leaks from the process line inside the sample hood, and (3) leaks from the process line outside the sample hood, which includes the piping in the sample cart. The number of MCOs processed per year, the probability of an operator error in properly installing the MCO valve operator, and the chance of a sampling line failure due to internal or external forces (SNF-4042) combine to place the annual frequency of this event into the anticipated category.

The unmitigated radiological dose to the onsite worker from the gaseous release accident is calculated to be less than the onsite risk evaluation guidelines for anticipated events and, therefore, no additional mitigative or preventive features are required.

The dominant defense-in-depth features of the CSB design to reduce doses associated with sampling system leak accidents are the HEPA filter train of the sample hood air system, which discharges to the CSB general exhaust system, and the HEPA filter on the sample line located on the sample hood. These filters will mitigate releases associated with breaches of the sample line inside and outside of the sample hood.

### A3.4.2.2.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls

The following safety SSCs and TSR controls are designated to mitigate the dose consequences of the bounding gaseous release accidents:

#### G1. Gaseous Release during MCO Sampling Operations

The release of MCO gases during the MCO sampling operation is due to a leak from the sampling system. Such a leak could result from operator error, the drop of objects onto the sampling/weld station, or the
sampling/weld station gantry crane, the MHM, or some other equipment moving horizontally into the sampling/weld station components and causing a shear of the piping or otherwise failing the sampling system pressure boundary. No specific safety features and controls are required to prevent or mitigate this event.

G2. Gaseous Release from Overpressurization. The release of MCO gases resulting from overpressurization of the MCO is caused by the inert gas system during reinerting of a monitored MCO after sampling. This accident is initiated by failure of the pressure regulator on the helium supply system while an MCO is in the sampling/weld station undergoing the MCO helium backfill operation. The following is the specific safety feature that prevents or mitigates this event:

- Safety-significant SSCs
  - Pressure relief device (rupture disk) — The rupture disk is provided to prevent releases from the sampling system when sampling the MCO. This pressure safety device will maintain pressure to less than design value (150 lb/in² gauge) when sampling the MCO.

In addition, the maintenance of gas cylinders is assumed to be controlled by the site/project maintenance program. This program will verify that the gas cylinders have cylinder valves that prevent excessive gas flow from the cylinders. As such, a TSR is not required.

The SSCs, controls, and defense-in-depth features designated to mitigate gaseous release accidents are summarized in Table A3-16. NRC important-to-safety category SSCs and defense-in-depth features also are included for each specific accident in Table 3-16.

G3. Gas Cylinder Impacts. A high-pressure gas cylinder is not expected to be propelled and impact the sample system and shear the piping because the gas cylinders used for MCO sampling and welding operations will be located outside the CSB. The only bottles available to create this hazard are those on the tube vent and purge cart and those brought into the facility for maintenance welding activities. Accidentally shearing the valve body from the cylinder, by dropping the cylinder or crashing something into the cylinder, would not result in the cylinder causing serious propulsion effects because the cylinder valve butt that remains in the neck of the bottle severely limits the flow rate of gas from the cylinder. Limiting the flow rate limits the thrust generated by the escaping gas such that the cylinder will not accelerate to a dangerous velocity. Removal of the valve in the neck of the pressurized cylinder is an extremely difficult task that could not be accomplished by one person without special equipment. The following is the specific safety feature that prevents or mitigates this event:

- Design feature
  - Maximum hole diameter in the butt of the valve on all gas cylinders — Limits the thrust produced by escaping gas to an amount that cannot accelerate the cylinder to dangerous velocity; this feature protects against cylinder propulsion caused by accidentally breaking off the cylinder valve.
### Table A3-16. Summary of Safety Features Required to Mitigate the Consequences of a Gaseous Release Design Basis Accident. (2 sheets)

<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designator*</th>
<th>General function</th>
<th>Safety feature and safety classification*</th>
<th>NRC ITS category*</th>
</tr>
</thead>
</table>
| G1. Gaseous release during MCO sampling operations | WS-F-02 WS-F-05 WS-G-03b WS-G-04b WS-G-06b WS-G-07b | Prevent equipment, moving horizontally, damaging the sampling system | Defense in depth:  
- MHM interlock (P10), sensors, and switches (MHM collision avoidance system)  
- Seismic restraints on the sampling/weld station gantry crane  
- Sampling/weld station gantry crane crash shield guard  
- MCO sampling piping confinement system  
- Sample hood and sample hood exhaust system  
- Sample line HEPA filter  
- Sample hood exhaust flow indicating device  
- Personnel are trained to procedures detailing the safe sequence of operations  
- The operators of the MHM are trained and qualified to perform their duties safely, which includes following procedures for safe operation  
- The MHM collision avoidance including guard rails  
- Personnel are trained to facility-specific procedures in the proper handling of the transportation cask, MCO, receiving crane, gantry, and MHM  
- The sampling/weld station gantry crane crash shield guard activates the MHM anticollision system to prevent collisions with the sample hood and sample lines  
- Lifting devices used at the CSB are designed to handle the loads they will carry  
- Regular maintenance is performed on the MHM to ensure it is in good working order | |
| G2. Gaseous release from overpressurization | WS-H-06a WS-H-07 WS-H-11 | Protect the sampling system piping and the MCO from internal pressure that exceeds design pressure | Safety-significant SSCs:  
- Helium supply system pressure relief device (rupture disk) | B* |
| | | | Defense in depth:  
- Pressure relief devices on the MCO sampling system and on the helium supply system  
- Helium supply system pressure regulator | |
## Table A3-16. Summary of Safety Features Required to Mitigate the Consequences of a Gaseous Release Design Basis Accident. (2 sheets)

<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designator</th>
<th>General function</th>
<th>Safety feature and safety classification</th>
<th>NRC ITS category</th>
</tr>
</thead>
<tbody>
<tr>
<td>cylinder</td>
<td></td>
<td>high-pressure</td>
<td>• Maximum hole diameter in the cylinder</td>
<td></td>
</tr>
<tr>
<td>impacts</td>
<td></td>
<td>gas cylinders</td>
<td>valves on all gas cylinders</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>from</td>
<td>Defense in depth:</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>damaging</td>
<td>• Operators are trained in the proper</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>sample lines</td>
<td>handling of compressed gas cylinders</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>according to established procedures</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Gas cylinders supplying the inert gas</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>system are located outside of the CSB</td>
<td></td>
</tr>
</tbody>
</table>

Note: Defense-in-depth features and safety-significant defense-in-depth features are not designated as NRC ITS.


*bSSCs are classified per their function in mitigating or preventing specific accidents. SSCs may have other classifications based on their functions in other events.


*There is only an impact to the MCO vessel in this accident — no offsite dose greater than 5 rem occurs and there is no criticality problem.

CSB = Canister Storage Building.
HEPA = high-efficiency particulate air (filter).
ITS = important to safety.
MCO = multi-canister overpack.
MHM = multi-canister overpack handling machine.
NA = not applicable to ITS category classification.
NRC = U.S. Nuclear Regulatory Commission.
SSC = structure, system, and component.
TSR = technical safety requirement.
A3.4.2.3 Multi-Canister Overpack Internal Hydrogen Deflagration. Three design basis events could result in the formation of flammable mixtures of hydrogen and oxygen within an MCO, which, if ignited, could result in a deflagration. The events are (1) radiolytic decomposition of the oxygen containing compounds, (2) introducing oxygen into the MCO during charging, and (3) ingress of oxygen following an MCO breach. The first two events, radiolytic decomposition and introduction of oxygen, have been evaluated further. The third event, ingress of oxygen, is similar to the second event; however, when comparing accident frequencies and consequences, the introduction of oxygen during charging is considered to present a substantially greater risk.

A3.4.2.3.1 Scenario Development. The following paragraphs describes the scenario development for each of the cases evaluated.

Radiolytic Decomposition. This scenario involves the radiolytic decomposition of the aluminum hydroxide, aluminum and iron hydrates, free water and water bound as hydrates in the uranium oxide and the potential for flammable mixtures of hydrogen and oxygen in an MCO. Of particular interest is the concentration of oxygen within the MCO. The detailed analysis of the potential for a flammable mixture within the MCO, and the minimum concentration of oxygen required, is contained in HNF-SD-SNF-TI-040.

To determine the potential for a flammable mixture within an MCO (i.e., oxygen concentration exceeding 4% [HNF-SNF-SD-TI-040]) two mechanisms were evaluated. The mechanisms included thermal decomposition, and radiolysis of aluminum hydroxide and uranium oxide hydrates. HNF-SD-SNF-TI-040 concludes that the aluminum hydroxide will only decompose by radiolysis based on CSB storage conditions (temperature). Over a 40-year period, only 4% of the aluminum hydroxide cladding material will decompose and about 10% of the particulate will decompose because of radiolysis. Uranium oxide hydrates will decompose thermally; however, the presence of water vapor limits thermal decomposition. The analysis contained in HNF-SNF-SD-TI-040 considers only radiolytic decomposition because radiolytic decomposition will generate more moles of gas (including oxygen) per mole of solid. Approximately 50% of the uranium oxide hydrate will decompose by radiolysis in a 40-year period.

Using the reference g(O2) values from HNF-SD-SNF-TI-015, the MCO oxygen concentration in the MCO remains below 4% after 40 years. The analysis of a zero scrap basket configuration (i.e., low reactive area) at various MCO power and water contents with nominal helium backfill does not result in oxygen concentrations above 4% over 40 years. All one and two scrap basket configurations also do not exceed 4% oxygen concentration over a 40-year period.

The conclusions reached in HNF-SNF-SD-TI-040 regarding radiolytic decomposition and the production of oxygen have been validated through literature reviews. From the data reviewed, approximately 50% of the uranium hydrate is expected to decompose over the 40-year storage time. Much of this decomposition results from alpha and beta radiolysis, which has been
projected to produce stoichiometric evolution of oxygen. The overall stoichiometry of radiolysis
does not produce a large increase in oxygen that must be incorporated in the crystal structure.
Therefore, it is concluded that the nonstoichiometric G-values for hydrated oxides can be
sustained throughout the 40-year storage period (SNF-3328).

The potential for a flammable mixture (i.e., oxygen concentration greater than 4%) using
reference \( g(O_2) \) values is considered beyond extremely unlikely because of the ability of the solid
materials to incorporate excess oxygen. Thus an internal deflagration and possible detonation
during the 40-year interim storage period are also considered beyond extremely unlikely.

Introduction of Oxygen into a Multi-Canister Overpack. This accident involves
insertion of inert gas contaminated with oxygen into an MCO being sampled. The cover cap
welding on selected MCOs will be delayed to allow for gas sampling. An MCO to be sampled is
moved to the sampling/weld station and a sample hood is placed over the MCO. Connections are
made to the MCO sample port and MCO gas is collected using the positive pressure of the MCO.
Inert gas is added to raise the pressure if necessary. Flammable mixtures of hydrogen and oxygen
can form in the MCO if the inert gas is contaminated with oxygen.

The following process conditions were inputs to the accident analysis:

- The MCO contains the maximum hydrogen inventory corresponding to reaction of all
  200 g of free water and half of the water bound as hydrates. The radiolytic
decomposition of water and aluminum hydroxide adds very little gas during the first
year after sealing and is not considered. The bounding MCO, which contains the
maximum hydrogen inventory, contains two scrap baskets and three fuel baskets.

- The pressure in the MCO after hydrogen combustion can be estimated from the
  adiabatic flame temperature of the reacting gases. Basically, an assumption is made
that 100% of the oxygen reacts to form water vapor. The heat of formation and the
specific heat capacity of the post-combustion gases determine the final temperature of
the gas mixture. The ideal gas law is then used to estimate the pressure due to the
combustion.

- An MCO helium inerting pressure of 1.1 atm is used in the analysis to maximize the
hydrogen concentration. The helium pressure is normally about 1.5 atm or more. A
contaminated helium supply at either pressure would lead to flammable mixtures.

The unmitigated scenario is brought to a stable state by allowing the MCO to depressurize
to atmospheric pressure. If the MCO is inside the cask, MHM, or the storage tube, these also are
allowed to depressurize to atmospheric pressure. The MCO is handled within recovery
operations under operational related procedures, with the preferred approach being to move the
MCO to the overpack storage tube for long-term observation and storage.
Mixtures of hydrogen in air are flammable in the range of 4% to 75% hydrogen by volume. The minimum oxygen concentration required for combustion is 4%. According to NUREG/CR-2726, *Light Water Reactor Hydrogen Manual*, high-pressure shock waves can be produced if the hydrogen concentration is between 18 vol% and 58 vol% in air.

The oxygen concentration in the MCO at the time of the accident depends on the oxygen concentration in the CSB helium system and on the amount of helium and hydrogen in the MCO before sampling. The helium inventory in the MCO can be estimated from the ideal gas law and the filling pressure at the CVDF, assuming a conservatively low MCO gas temperature due to the operation of the tempered water system.

Over a period of weeks, the available water in the MCO would react with uranium metal and uranium hydride to form uranium oxide and hydrogen provided there is no oxygen in the MCO. This assumption is realistic because oxygen reacts with uranium metal more readily than water. The water inventory has two parts: (1) the 200 g of free water found in cracks and crevices and (2) the water contained as a hydrate in the oxide films and aluminum hydroxide. For an MCO with two scrap baskets and three fuel baskets, the maximum hydrate water is 1,190 g (HNF-SD-SNF-TI-015). Because the uranium hydrate water loss is temperature dependent (HNF-1523 [Rev. 1]), it is assumed that half of this hydrate is available to become vapor and reacts with the uranium fuel to form hydrogen (see discussion and table in Section A3.4.2.5.1). The total water available to form hydrogen for temperatures less than 100 °C is 800 g, leading to the production of 124 g of hydrogen. This includes the production of hydrogen from hydride decomposition. To maximize the hydrogen concentration, the helium fill pressure at the CVDF is assumed to be 1.1 atm instead of 1.5 atm. This yields a concentration of 73% hydrogen and 27% helium in the MCO on receipt at the CSB. The MCO gas temperature at the CSB is assumed to be 50 °C (122 °F).

When the MCO is moved from the storage tube to the sampling station for analysis, a sampling hood is placed over it. The hood is connected to a HEPA-filtered exhaust system. A sampling line is connected to the MCO, and a gas sample is collected using the pressure of the MCO. If the MCO were inadvertently depressurized, a recharge sequence would be initiated by the operator to add helium and restore the MCO pressure.

When the MCO is depressurized, the relative amounts of helium and hydrogen remain unchanged. The gas mixture is still 27% helium and 73% hydrogen. At a pressure of 1.0 atm and a temperature of 50 °C (122 °F), the MCO inventory is 18.9 gmoles total, or 13.9 gmoles hydrogen and 5.0 gmoles helium.

The action of adding a helium-oxygen mixture to the MCO increases the pressure in the MCO to 1.27 atm. The amount of gas added is calculated from the ideal gas law to be 5.1 gmoles. The final total number of moles of gas in the MCO is 24.0 gmoles. The hydrogen concentration in the MCO is diluted from 73% to 58%. The oxygen concentration in the MCO depends on the oxygen concentration in the added gas. From the ideal gas law, the final oxygen concentration in the MCO is 21.3% of its concentration in the added helium-oxygen mixture.
If the added mixture were 60% oxygen, then the final oxygen concentration in the MCO would be 12.8%.

For a flammable mixture to occur at atmospheric pressure, the oxygen concentration must exceed 4% in the MCO. Thus, the contaminated inert gas has an oxygen concentration greater than 23%. For a mixture in the MCO to support a detonation with shock waves, the oxygen concentration in the MCO must be at least 9%. The flammable mixture in the MCO does not support damaging shock waves until the oxygen concentration in the added gas exceeds 42%.

If the contaminated inert gas added to the MCO is 60% oxygen, then the final composition of the MCO gas is 7.1 gmoles helium, 13.9 gmoles hydrogen, and 3.0 gmoles oxygen. The MCO pressure at this time is 1.27 atm. The MCO gas temperature is 50 °C (122 °F). The complete combustion of this mixture yields an energy release of 355,000 calories with a final theoretical temperature of the gas mixture of about 2,790 °C (5,054 °F). The corresponding pressure in the MCO is 10.5 atm. The oxygen concentration of the contaminated gas is high enough that there is a potential for detonation shock waves to form. The pressure from a detonation could be as much as two times higher in localized areas than the pressure from the burn alone. Before welding, the MCO rated pressure is 150 lb/in² (SNF-3328); the MCO is not expected to be damaged by the worst-case deflagration. An MCO with mechanical closure is stable to an internal loading of 340 lb/in², and after welding, rated pressure is 450 lb/in² (HNF-SD-SNF-SARR-005). In addition, the sample line and connection are expected to withstand these pressures. However, the sample line is assumed to fail and release gas and particulate from the MCO into the environment.

A3.4.2.3.2 Source Term Analysis. For the internal hydrogen deflagration, the MAR (particulate available for release) depends on the amount of particulate generated between the time the fuel is washed at the K Basins and the time of the accident. The Technical Databook (HNF-SD-SNF-TI-015) provides a bounding estimate of the MAR for any time during the 40-year life of the facility. The Technical Databook (HNF-SD-SNF-TI-015) identifies the safety basis or bounding value for the mass of particulate as 34.0 kg UO₂, which contains 30.0 kg of uranium. Because of the conservative methodology used to calculate the bounding value, 30 kg is considered representative of the 99th percentile. The Technical Databook also identifies a design or nominal value of 2.1 kg UO₂, which contains 1.85 kg of uranium.

The quantity of particulate actually released during an accident depends on the initial aerosol concentration, which includes that generated by accident forces, and subsequent particulate entrainment. This blowdown scenario would feature films of particles rather than deep powder beds. In such a situation, there would still be gas flow through the bed during depressurization, but the local velocity is simply not sufficient for entrainment (IDCOR 1984). The pressurized powder venting source term is not applicable, and a source term based upon aerodynamic entrainment, as described in Section A3.4.2.1.2, is used.

Based on the data reported in DOE-HDBK-3010-94, a bounding (95th percentile) ARR of $4 \times 10^5$/h and an RF of 1.0 were selected as bounding for the blowdown. The contents of the MCO are intact fuel elements tightly packed in fuel baskets and pieces of fuel elements housed in...
the scrap baskets. Particulate matter swept upward by streams of flowing gas within the MCO
must take a tortuous path through the MCO and through the MCO shield plug to exit the MCO.
For shielded powder, where the aerodynamic stresses are reduced by debris or exposure to static
conditions, DOE-HDBK-3010-94 recommends an ARR of $4 \times 10^6$/h and an RF of 0.2 for
powder under debris. These values were selected as nominal (50th percentile).

The initial pressure in the MCO at the start of the blowdown determines the fraction of the
suspended particulate that is released. The range of pressure for an internal hydrogen deflagration
varies with the composition of the initial flammable mixture. According to NUREG/CR-2726, the
range of pressure from a hydrogen burn would be from 3 to 12 times the initial pressure. For the
calculation of the source term, 5 atm is assumed to be the nominal pressure (50th percentile) with
12 atm used as the 95th percentile pressure. This results in a nominal (50th percentile) LPF of 0.8
for the MCO and a 95th percentile LPF of 0.92.

The pressurized release from the MCO following the internal deflagration would flow most
likely through a leak path with a flow area greater than or equal to the flow area through an MCO
shield plug port. The duration of the blowdown depends on the flow area of the leak and the
resultant pressure following the burn. The MCO depressurizes to atmospheric pressure in less
than 1 minute, even for an MCO pressure of about 150 lb/in$^2$ gauge and a 0.25-in. hole. The
MCO depressurizes in a much shorter time for lower pressures and larger leak paths. For the
calculation of the source term, a nominal (50th percentile) value of 10 seconds and a bounding
(95th percentile) time of 60 seconds were selected.

The material released by aerosol entrainment can be calculated using the methodology
described in Section A3.4.1.3 and the bounding and nominal values identified above. The
bounding (95th percentile) value for the material released by aerodynamic entrainment is
$2.0 \times 10^3$ g, as shown in Table A3-17.

The blast and shock from the hydrogen deflagration would suspend some of the particulate
within the MCO. The powder at rest in the MCO could be ejected into the gas volume by the
response of the underlying solid MCO and fuel substrate to the vibration and jolting induced by
deflagration. According to DOE-HDBK-3010-94, the value of the ARF under such
circumstances should exceed the value of the ARF for aerodynamic suspension alone but be less
than the value of the ARF for the free-fall of powder. The powder undergoing vibration shock is
bounced into the gas while subject to the same gas velocities as those for aerodynamic
entrainment. Based on the discussion in DOE-HDBK-3010-94, a bounding (95th percentile) ARF
of $1 \times 10^3$ and an RF of 1.0 were chosen for the suspension of powder-like surface contamination
by blast and shock vibration. A nominal (50th percentile) value of $4 \times 10^4$ was chosen for the
ARF and a nominal (50th percentile) value of 0.2 was chosen for the RF based on the evaluation
of the discussion in DOE-HDBK-3010-94.
Table A3-17. Aerodynamic Entrainment Source Term for the Internal Deflagration Accident.

<table>
<thead>
<tr>
<th>Nominal</th>
<th>Bounding</th>
<th>EF</th>
<th>Percentile</th>
<th>SNV</th>
<th>In(EF)/SNV</th>
</tr>
</thead>
<tbody>
<tr>
<td>MAR</td>
<td>1.85 kg</td>
<td>30 kg</td>
<td>16.22</td>
<td>99</td>
<td>2.326</td>
</tr>
<tr>
<td>ARR</td>
<td>4.00 E-06/h</td>
<td>4.00 E-05/h</td>
<td>10.00</td>
<td>95</td>
<td>1.645</td>
</tr>
<tr>
<td>Time</td>
<td>10 s</td>
<td>60 s</td>
<td>6.00</td>
<td>95</td>
<td>1.645</td>
</tr>
<tr>
<td>RF</td>
<td>0.20</td>
<td>1.00</td>
<td>5.00</td>
<td>95</td>
<td>1.645</td>
</tr>
</tbody>
</table>

M = 4.1 E-06 g\(^a\) 2.0 E-04 g\(^b\) 48\(^c\) 95 1.645 2.353

Note: The 1.645 is the standard normal variable corresponding to the 95% upper confidence limit. For the 99% upper confidence limit, the corresponding value is 2.326.

\(^a\)Nominal_{\text{overall}} = MAR_{\text{nominal}} \times ARR_{\text{nominal}} \times RF_{\text{nominal}} \times \text{time}_{\text{nominal}}

\(^b\)Bounding_{\text{overall}} = (EF_{\text{overall}}) (\text{nominal}_{\text{overall}} M)

\(^c\)\text{EF}_{\text{overall}} = \exp \left( \text{SNV}_{95\%} \left( \sum_{i} \left( \frac{\ln(EF)_{i}}{\text{SNV}_{i}} \right)^2 \right)^{\frac{1}{2}} \right)

ARR = airborne release rate.
EF = error factor (i.e., bounding value divided by nominal value).
ln(EF) = natural log of EF.
M = mass of material released.
MAR = material at risk.
RF = respirable fraction.
SNV = standard normal variable.

The material released by shock impact can be calculated using the methodology described in Section A3.4.1.3 and the bounding and nominal values identified above. The bounding (95th percentile) value for the material released by shock impact is 1.8 g, as shown in Table A3-18.

The bounding (95th percentile) source term for the internal hydrogen deflagration is 1.8 g, which is the sum of 1.8 g from vibration and shock and 2.0 \times 10^4 g from aerodynamic entrainment.
Table A3-18. Initial Suspended Material Source Term for the Internal Deflagration Accident.

<table>
<thead>
<tr>
<th></th>
<th>Nominal</th>
<th>Bounding</th>
<th>EF</th>
<th>Percentile</th>
<th>SNV</th>
<th>ln(EF)/SNV</th>
</tr>
</thead>
<tbody>
<tr>
<td>MAR</td>
<td>1.85 kg</td>
<td>30 kg</td>
<td>16.22</td>
<td>99</td>
<td>2.326</td>
<td>1.198</td>
</tr>
<tr>
<td>ARF</td>
<td>4.00 E-04</td>
<td>1.00 E-03</td>
<td>2.50</td>
<td>95</td>
<td>1.645</td>
<td>0.557</td>
</tr>
<tr>
<td>RF</td>
<td>0.20</td>
<td>1.00</td>
<td>5.00</td>
<td>95</td>
<td>1.645</td>
<td>0.978</td>
</tr>
<tr>
<td>LPF_{MCO}</td>
<td>0.80</td>
<td>0.92</td>
<td>1.15</td>
<td>95</td>
<td>1.645</td>
<td>0.083</td>
</tr>
</tbody>
</table>

Note: The 1.645 is the standard normal variable corresponding to the 95% upper confidence limit. For the 99% upper confidence limit, the corresponding value is 2.326.

\[ *LPF_{MCO} = \frac{(MCO \text{ pressure} - 1 \text{ atm})}{(MCO \text{ pressure})}. \]

At a nominal MCO pressure of 5 atm, the \( *LPF_{MCO} \) is 0.80, and at a bounding MCO pressure of 12 atm, the \( *LPF_{MCO} \) is 0.92.

\[ {\text{Bounding}_{\text{overall}}} M = \text{MAR}_{\text{overall}} \times \text{ARF}_{\text{overall}} \times \text{RF}_{\text{overall}} \times *LPF_{\text{MCO}}. \]

\[ \text{ARF} = \text{airborne release fraction}. \]
\[ \text{EF} = \text{error factor (i.e., bounding value divided by nominal value)}. \]
\[ \text{LPF} = \text{leak path factor}. \]
\[ \text{ln}(\text{EF}) = \text{natural log of EF}. \]
\[ M = \text{mass of material released}. \]
\[ \text{MAR} = \text{material at risk}. \]
\[ \text{MCO} = \text{multi-canister overpack}. \]
\[ \text{RF} = \text{respirable fraction}. \]
\[ \text{SNV} = \text{standard normal variable}. \]

**A3.4.2.3.3 Consequence Analysis.** The dose calculation equation and data from Section 3.4.1 of the SNF Project FSAR are used to calculate the dose to the onsite receptor.

\[ D_{\text{onsite}} = M \times \frac{X}{Q'} \times BR \times UD \times *LPF_{\text{building}} \]
\[ = (1.8 \text{ g U})(1.14 \times 10^{-2} \text{ s/m}^3)(3.33 \times 10^{-4} \text{ m}^3/\text{s})(4.38 \times 10^5 \text{ rem/g U})(1.0) \]
\[ = 3.0 \text{ rem (0.03 Sv)}. \]
where

\[ D_{\text{ onsite}} \] = committed effective dose equivalent (rem)

\[ M \] = mass of respirable airborne material released (g U)

\[ \chi/Q \] = time-integrated atmospheric transport factor (s/m³)

\[ BR \] = breathing rate (m³/s)

\[ UD \] = dose per unit mass of uranium (rem/g U)

\[ \text{LPF}_{\text{building}} \] = leak path factor from building.

The dose consequences at the remaining receptor sites are calculated in the same manner and are shown in Table A3-19.

Table A3-19. Dose Calculation Summary for a Bounding Internal Hydrogen Deflagration at the Sampling/Weld Station.

<table>
<thead>
<tr>
<th>Receptor location</th>
<th>Duration (hours)</th>
<th>Unmitigated dose(^a) rem (Sv)</th>
<th>Evaluation guideline(^b)/release limits rem (Sv)</th>
<th>Mitigated dose rem (Sv)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Onsite (with building effects) (100 m E)</td>
<td>&lt;1</td>
<td>3.0 (3.0 E-02)</td>
<td>10 (1.0 E-01)</td>
<td>--</td>
</tr>
<tr>
<td>Highway 240° (9,280 m W)</td>
<td>&lt;1</td>
<td>6.1 E-03 (6.1 E-5)</td>
<td>--</td>
<td>--</td>
</tr>
<tr>
<td>Hanford Site boundary (17,390 m E)</td>
<td>&lt;1</td>
<td>3.4 E-03 (3.4 E-05)</td>
<td>5.0 (5.0 E-02)</td>
<td>--</td>
</tr>
</tbody>
</table>

\(^a\)Fifty-year committed effective dose equivalent.

\(^b\)Evaluation guideline for onsite (100 m) receptor only.

\(^c\)Unmitigated frequency for this event is unlikely (\(>10^4\) to \(\leq 10^2\) per year)

\(^d\)Provided for information only.

A3.4.2.3.4 Comparison to Guidelines. The unmitigated radiological offsite dose for this event is below offsite release limits, and the unmitigated onsite dose does not exceed onsite evaluation guidelines for an unlikely event.

A3.4.2.3.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls. No safety-class SSCs are required to prevent the hydrogen deflagration DBA or to mitigate its dose consequences. Under normal operating conditions, there is no anticipated internal accumulation of flammable mixtures of hydrogen and oxygen.

11. Oxygen Buildup from Radiolysis. A hydrogen deflagration (SA-J-06a, OA-J-06a, WS-J-06a) during cask handling follows from excessive gas buildup in the MCO caused by radiolysis. It has been shown that the bounding case MCO will not be able to generate flammable gas mixtures even after 40 years of storage. Somewhat larger inventories of aluminum hydroxide
remaining in the MCO after washing at K Basins could lead to flammable mixtures after many
years of storage. For short durations (i.e., prior to welding), the MCO design pressure
(150 lb/in²) will not be exceeded in the worst-case deflagration or gas buildup. The MCO with a
mechanical seal will not fail (i.e., is stable) up to a pressure of 340 lb/in²
(HNF-SD-SNF-SARR-005). During long-term storage, combustion of these mixtures will not
exceed the 450 lb/in² design pressure of the welded MCO. To exceed this pressure limit, the key
interface performance assumptions identified in Section A3.4.3.1 must be exceeded. Exceeding
these key assumptions is discussed in Section A3.4.1. The following control prevents this event:

- TSR

  - Verify the proper processing steps and conditions have been carried out at
    CVDF and K Basins before accepting an MCO at CSB.

12. **Oxygen Used as a Purge Gas.** Good operating practices during the sampling process
will not significantly depressurize the MCO. As little sample gas as possible should be taken from
the MCO during sampling. Using a sampling operation that minimizes the need to refill the MCO
with inert gas reduces the frequency with which this accident could occur. This sampling
operation should be considered defense in depth for prevention of flammable gas mixtures within
the MCO at the sampling station.

The SSCs and TSR controls designated to protect the assumptions of the analysis for MCO
internal hydrogen deflagration accidents are summarized in Table A3-20. Defense-in-depth
features also are included in Table A3-20.
<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designator</th>
<th>General function</th>
<th>Safety features and safety classification</th>
<th>NRC ITS category</th>
</tr>
</thead>
<tbody>
<tr>
<td>11. Oxygen buildup from radiolysis</td>
<td>SA-J-06a, OA-J-06a, WS-J-06a</td>
<td></td>
<td>TSR: • Verify the proper processing steps and conditions have been carried out at CVDF and K Basins before accepting an MCO at CSB</td>
<td></td>
</tr>
<tr>
<td>12. Oxygen used as a purge gas</td>
<td>WS-H-06b, OA-I-06c</td>
<td>Prevent oxygen from entering the inert gas system</td>
<td>Defense in depth: • Quality assurance program verifies vendor certification of helium minimum purity of 99% verified on receipt at the CSB for any inert gas cylinders used to supply the inert gas system. • Supply lines inerted after they have been depressurized before using inert gas system</td>
<td></td>
</tr>
</tbody>
</table>

Note: Defense-in-depth features and safety-significant defense-in-depth features are not designated as NRC ITS.


*SSCs are classified per their function in mitigating or preventing specific accidents. SSCs may have other classifications based on their functions in other events.

CSB = Canister Storage Building.
CVDF = Cold Vacuum Drying Facility.
ITS = important to safety.
MCO = multi-canister overpack.
NRC = U.S. Nuclear Regulatory Commission.
SSC = structure, system, and component.
TSR = technical safety requirement.
A3.4.2.4 Multi-Canister Overpack External Hydrogen Deflagration. Based on the transport
time restrictions identified in HNF-SD-TP-SARP-017, insufficient pressure will be generated
during the transport of the MCO from the CVDF to the CSB to fail the mechanical seal and
release gases to the internals of the shipping cask. In addition, HNF-SD-TP-SARP-017 requires
the cask to be vented to the CSB facility if the cask–MCO is stored at the CSB for an extended
period. Therefore, the event of concern is a hydrogen release during sampling activities because
of (1) releases from an MCO during sampling activities and (2) an inadequately sealed MCO
shipped from the CVDF to the CSB.

Particulate releases from an MCO during sampling activities have been previously
evaluated (see Section A3.4.2.3). However, the release of hydrogen during sampling activities
assumes that a gaseous release (primarily hydrogen) occurs and that the hydrogen mixes with the
oxygen in the CSB facility creating a flammable mixture that is ignited. External hydrogen events
in the MHM and the storage tube have the same risk ranking as the external hydrogen
deflagration at the sampling/weld station (HNF-SD-SNF-HIE-001), but the deflagration at the
sampling/weld station was chosen as the DBA because it has a higher potential to impact worker
safety than an external hydrogen event in the MHM or storage tube. Storage of the MCO in a
cask at the CSB is discussed in SNF-3328, including consideration of a hydrogen deflagration.

A3.4.2.4.1 Scenario Development. The DBA scenario identified is the release of
flammable gases from an MCO during sampling activities. The cover cap welding on selected
MCOs will be delayed for approximately one year to allow for gas sampling at the CSB. An
MCO to be sampled is moved to the sampling/weld station, and a sample hood is placed over the
MCO. Connections are made to the MCO sample port, and MCO gas is collected using the
positive pressure of the MCO. If the connection to the MCO fails and discharges MCO gas into
the sample hood, a flammable mixture of hydrogen and air could be formed in the hood.
Personnel are normally near the hood during this time, and it is possible that a deflagration in the
sample hood would cause a serious injury to the nearby operator opening the MCO port valve. In
addition, the deflagration in the sample hood causes additional damage to the sample line and
connector, so a gaseous release of particulate from the MCO also may occur.

The following planned process conditions and assumptions are used in the accident analysis
calculations.

- The MCO contains the maximum hydrogen inventory corresponding to reaction of
  all the residual free water and half of the water bound as hydrates. The radiolytic
decomposition of water and aluminum hydroxide adds very little gas and will not be
  considered. The MCO yielding the maximum hydrogen inventory contains two
  scrap baskets and three fuel baskets.

- The approximate volume of the sample hood is 600 L. The dimensions and volume
  of the sample hood are not critical to the accident analysis conclusions. Larger or
  smaller volumes can be used to give the same flammable concentration by adjusting
  the assumed MCO leak rate.
The sample hood exhaust heating, ventilation, and air conditioning (HVAC) system is operating normally, but at its lowest flow rate, 100 ft³/min. The pressure in the MCO is relieved at the rate of 42 ft³/min due to a failure of the sample line connection.

The bounding pressure in the sample hood can be estimated from the adiabatic flame temperature of the reacting gases. Basically, one assumes 100% of the hydrogen reacts to form water vapor. The heat of formation and the heat capacity of the post-combustion gases determine the final temperature of the gas mixture. The ideal gas law is then used to estimate the pressure due to the combustion.

The MCO helium inerting pressure before it is sealed at the CVDF is 1.5 atm.

An ignition source is present in the sample hood and causes a deflagration of the hydrogen, helium, and air mixture in the hood.

The sample line connection is not properly made, so MCO gases leak from the connection and the resulting deflagration leads to a pressurized release of gas and particulate in the MCO.

The particulate not removed from the fuel by washing at the K Basins will not be dislodged and released by the pressurized gas release.

The unmitigated scenario is brought to a stable state by allowing the MCO to vent any gas continuing to be generated. If the MCO is inside the cask, MHM, or the storage tube, these also are allowed to continue venting. A confinement service tent is used to cover the MCO and its vehicle, if appropriate, until the continued gas generation subsides to low levels a few hours after the event. Any contamination is cleaned consistent with radiation control procedures. The MCO is handled within recovery operations under emergency response procedures, with the preferred approach being to move the MCO to the overpack storage tube for short-term observation and storage.

The analysis of the hydrogen deflagration at the sampling/weld station focuses on the hydrogen concentration in the sample hood. Mixtures of hydrogen in air are flammable in the range of 4% to 75% hydrogen by volume. High-pressure shock waves can be produced if the hydrogen concentration is between 18 vol% and 58 vol% in air (NUREG/CR-2726).

The hydrogen concentration in the sample hood at the time of the accident depends on the hydrogen concentration in the MCO, the sample line leak rate, and the hood ventilation rate. The helium inventory in the MCO can be estimated from the ideal gas law and the filling pressure at the CVDF, assuming a conservatively low MCO gas temperature due to the operation of the tempered water system.
Over a period of weeks, the available water in the MCO reacts with uranium metal and uranium hydride to form uranium oxide and hydrogen, provided there is no oxygen in the MCO. Typical exposed fuel surface areas are large enough to keep the oxygen content low. The bounding water inventory has two parts: (1) the 200 g of free water found in cracks and crevices and (2) the water contained in the oxide films as a hydrate and aluminum hydroxide. For an MCO with two scrap baskets and three fuel baskets, the maximum hydrate water is 1,190 g (HNF-SD-SNF-TI-015). Because the uranium hydrate can lose water rapidly (HNF-1523 [Rev. 1]), it will be assumed that half of this hydrate is available to become vapor and will react with the uranium fuel to form hydrogen. The total water available to form hydrogen at temperatures of less than 100 °C is 800 g, leading to the production of 124 g of hydrogen.

At the time of the accident, the gas mixture in the MCO is 67% hydrogen and 33% helium (initial condition assuming a fill pressure of 1.5 atm at the CVDF). At an MCO gas temperature of 75 °C (167 °F), the pressure of the helium and hydrogen mixture in the MCO is 63 lb/in² gauge.

When the MCO is moved from the storage tube to the sampling/weld station for analysis, a sampling hood is placed over it. The hood is connected to a HEPA-filtered exhaust system, and the flow rate through the hood is between 100 and 250 ft³/min. A sampling line is connected to the MCO with a port valve operator, and a gas sample is collected using the internal gas pressure of the MCO. If the sample connection fails or the valve operator is not fully installed, hydrogen and helium in the MCO would leak into the sample hood and its exhaust system.

Assuming a sample line failure occurs and the MCO discharges gas into the sample hood at the rate of 42 ft³/min, the hydrogen concentration quickly becomes flammable inside the sampling hood. Assuming the MCO gas mixes thoroughly with the air in the hood, only 2 seconds are needed for the hydrogen concentration to exceed the 4% hydrogen lower flammability limit. The peak concentration is 17% hydrogen after 27 seconds.

If the hydrogen concentration in the sample hood is 10% at the time of the deflagration, the sample hood would contain 4.9 g hydrogen. The adiabatic flame temperature would be 1,370 K, which leads to a maximum pressure of 49 lb/in² gauge. This pressure is large enough to damage the hood and injure any personnel nearby. The potential for significant personnel injury exists because an operator is normally located in front of the sample hood when opening and closing the MCO port valve.

A3.4.2.4.2 Source Term Analysis. In addition to causing personnel injury, the hydrogen deflagration in the sample hood can damage the sample line, resulting in depressurization of the MCO. Radioactive contamination in the MCO and the sample hood HEPA filter can be released to the environment. The primary sources of radioactivity are (1) the HEPA filter near the sample hood, (2) the particulate matter suspended inside the MCO by the deflagration, and (3) the entrained particulate matter resuspended and carried out with the gases leaving the MCO.
The MAR (particulate available for release) depends on the amount of particulate generated between the time the fuel is washed at the K Basins and the time of the accident. The Technical Databook (HNF-SD-SNF-TI-015) provides a bounding estimate of the MAR for any time during the 40-year life of the facility. The Technical Databook (HNF-SD-SNF-TI-015) identifies the safety basis or bounding value for the mass of particulate as 34.0 kg UO₂, which contains 30.0 kg of uranium. Because of the conservative methodology used to calculate the bounding value, 30 kg is considered representative of the 99th percentile. The Technical Databook (HNF-SD-SNF-TI-015) also identifies a design or nominal (50th percentile) value of 2.1 kg UO₂, which contains 1.85 kg of uranium.

In this accident, the MCO would blow down through the damaged sample line to relieve the pressure inside the MCO. However, such a blowdown scenario would feature films of particles rather than deep powder beds. In such a situation, there would still be gas flow through the bed during depressurization, but the local velocity is simply not sufficient for entrainment (IDCOR 1984). The pressurized powder venting source term is not applicable, and a source term based upon aerodynamic entrainment, as described in Section A3.4.2.1.2, is used.

Based on the data reported in DOE-HDBK-3010-94, a bounding (95th percentile) ARR of $4 \times 10^9$/h and an RF of 1.0 were selected. The contents of the MCO are intact fuel elements tightly packed in fuel baskets and pieces of fuel elements housed in the scrap baskets. Particulate matter swept upward by streams of flowing gas within the MCO must take a tortuous path through the MCO and through the MCO shield plug to exit the MCO. For shielded powder, where the aerodynamic stresses are reduced by debris or exposure to static conditions, DOE-HDBK-3010-94 recommends an ARR of $4 \times 10^{-6}$/h and an RF of 0.2 for powder under debris. These values were selected as nominal (50th percentile).

The pressurized release from the MCO following the external deflagration would flow mostly likely through a leak path with a flow area greater than or equal to the flow area through an MCO shield plug port. The duration of the blowdown depends on the flow area of the leak and the resultant pressure following the burn. The MCO depressurizes to atmospheric pressure in less than 1 minute, even for an MCO pressure of about 150 lb/in² gauge and a 0.25-in. hole. The MCO depressurizes in a much shorter time for lower pressures and larger leak paths. For the calculation of the source term, a nominal (50th percentile) value of 10 seconds and a bounding (95th percentile) time of 60 seconds were selected.

The material released by aerosol entrainment can be calculated using the methodology described in Section A3.4.1.3 and the bounding and nominal values identified above. The bounding (95th percentile) value for the material released by aerodynamic entrainment is $2 \times 10^4$ g, as shown in Table A3-21.
Table A3-21. Aerodynamic Entrainment Source Term for the External Hydrogen Deflagration Accident.

<table>
<thead>
<tr>
<th></th>
<th>Nominal</th>
<th>Bounding</th>
<th>EF</th>
<th>Percentile</th>
<th>SNV</th>
<th>ln(EF)/SNV</th>
</tr>
</thead>
<tbody>
<tr>
<td>MAR</td>
<td>1.85 kg</td>
<td>30 kg</td>
<td>16.22</td>
<td>99</td>
<td>2.326</td>
<td>1.198</td>
</tr>
<tr>
<td>ARR</td>
<td>4.00 E-06/h</td>
<td>4.00 E-05/h</td>
<td>10.00</td>
<td>95</td>
<td>1.645</td>
<td>1.400</td>
</tr>
<tr>
<td>Time</td>
<td>10 s</td>
<td>60 s</td>
<td>6.00</td>
<td>95</td>
<td>1.645</td>
<td>1.089</td>
</tr>
<tr>
<td>RF</td>
<td>0.20</td>
<td>1.00</td>
<td>5.00</td>
<td>95</td>
<td>1.645</td>
<td>0.978</td>
</tr>
</tbody>
</table>

M = $4.1 \times 10^{-6}$ g$^{a}$, $2.0 \times 10^{-4}$ g$^{b}$, 48$^{c}$, 95, 1.645, 2.353

Note: The 1.645 is the standard normal variable corresponding to the 95% upper confidence limit. For the 99% upper confidence limit, the corresponding value is 2.326.

$a$Nominal overall $M = MAR_{\text{nominal}} \times ARR_{\text{nominal}} \times RF_{\text{nominal}} \times \text{time}_{\text{nominal}}$

$b$Bounding overall $M = (EF_{\text{overall}}) (\text{nominal}_{\text{overall}} M)$

$\text{ARR} = \text{airborne release rate.}$

$\text{EF} = \text{error factor (i.e., bounding value divided by nominal value).}$

$\text{ln(EF)} = \text{natural log of EF.}$

$M = \text{mass of material released.}$

$\text{MAR} = \text{material at risk.}$

$\text{RF} = \text{respirable fraction.}$

$\text{SNV} = \text{standard normal variable.}$

A blast and shock could suspend some of the particulate within the MCO. The powder at rest in the MCO could be ejected into the gas volume by the response of the underlying solid MCO and fuel substrate to the vibration and jolting induced by deflagration. According to DOE-HDBK-3010-94, the value of the ARF under such circumstances should exceed the value of the ARF for aerodynamic suspension alone but be less than the value of the ARF for the free-fall of powder. The powder undergoing vibration shock is bounced into the gas while subject to the same gas velocities as those for aerodynamic entrainment. Based on the discussion in DOE-HDBK-3010-94, a bounding (95th percentile) ARF of $1 \times 10^3$ and an RF of 1.0 were chosen for the suspension of powder-like surface contamination by blast and shock vibration. A nominal (50th percentile) value of $1 \times 10^4$ was chosen for the ARF and a nominal (50th percentile) value of 0.2 was chosen for the RF based on the evaluation of the discussion in DOE-HDBK-3010-94.

As in the case of the gaseous release event, there is no clear dominating mechanism identified in this accident to provide the shock and impact to suspend particulate within the MCO.
However, normal handling of the MCO by the MHM during transport from the storage tube to
the sampling/weld station will produce some vibration and jostling of the fuel and suspend some
material. For the purposes of this analysis, it is assumed that nominally 1% of the impact
suspension concentration exists within the MCO at the start of sampling. Therefore a value of 1%
is assumed for the 50th percentile. In addition, a value of 10% is assumed for the 95th percentile.
The initial pressure in the MCO at the start of the blowdown determines the fraction of the
suspended particulate that is released from the MCO. The range of MCO pressure for this
scenario varies from a nominal (50th percentile) pressure of 2.0 atm to a maximum pressure of
6.0 atm, which is assumed to be the 95th percentile pressure. These pressures equate to a
nominal (50th percentile) LPF of 0.5 for the MCO and a 95th percentile LPF of 0.83.

The material released by shock impact can be calculated using the methodology described
in Section A3.4.1.3 and the bounding and nominal values identified above. The bounding
(95th percentile) value for the material released by shock impact is $2.7 \times 10^{-2}$ g, as shown in
Table A3-22.

The external hydrogen deflagration also has the potential to damage the HEPA filters in the
sample line and sample hood exhaust system. For the HEPA filter blasts, the bounding release
fraction is 0.01 with an RF of 1.0 (DOE-HDBK-3010-94, Section 5.2.2.2). The canister HEPA
filter attached to the sample hood is assumed to contain 20 g of fuel (SNF-3328). The amount of
fuel released from the HEPA filter by the blast of the hydrogen deflagration is 0.2 g.

The bounding (95th percentile) source term for the external hydrogen deflagration is
$0.23$ g, which is the sum of $2.0 \times 10^{-2}$ g from vibration and shock and $2.0 \times 10^{-4}$ g from
aerodynamic entrainment, plus an additional 0.2 g from the damage to the HEPA filter.

**A3.4.2.4.3 Consequence Analysis.** The dose calculation equation and data from
Section 3.4.1 of the SNF Project FSAR are used to calculate the dose to the onsite receptor.

\[
D_{\text{onsite}} = (M_{\text{MCO}} + M_{\text{HEPA}}) \times \frac{\chi}{Q'} \times BR \times UD \times LPF_{\text{building}}
\]

\[
= (0.23 \text{ g U})(1.14 \times 10^{-2} \text{ s/m}^{3})(3.33 \times 10^{-4} \text{ m}^{3}/\text{s})(4.38 \times 10^{5} \text{ rem/g U})(1.0)
\]

\[
= 0.38 \text{ rem (3.8 } \times 10^{-3} \text{ Sv)} .
\]

where

- $D_{\text{onsite}}$ = committed effective dose equivalent (rem)
- $M$ = mass of respirable airborne material released (g U)
- $\chi/Q'$ = time-integrated atmospheric transport factor (s/m$^3$)
- $BR$ = breathing rate (m$^3$/s)
- $UD$ = dose per unit mass of uranium (rem/g U)
- $LPF_{\text{building}}$ = leak path factor from building.
### Table A3-22. Initial Suspended Material Source Term for the External Deflagration Accident.

<table>
<thead>
<tr>
<th>Source Term</th>
<th>Nominal</th>
<th>Bounding</th>
<th>EF</th>
<th>Percentile</th>
<th>SNV</th>
<th>ln(EF)/SNV</th>
</tr>
</thead>
<tbody>
<tr>
<td>MAR</td>
<td>1.85 kg</td>
<td>30 kg</td>
<td>16.22</td>
<td>99</td>
<td>2.326</td>
<td>1.198</td>
</tr>
<tr>
<td>ARF</td>
<td>4.00 E-04</td>
<td>1.00 E-03</td>
<td>2.50</td>
<td>95</td>
<td>1.645</td>
<td>0.557</td>
</tr>
<tr>
<td>RF</td>
<td>0.20</td>
<td>1.00</td>
<td>5.00</td>
<td>95</td>
<td>1.645</td>
<td>0.978</td>
</tr>
<tr>
<td>Settling</td>
<td>1%</td>
<td>10%</td>
<td>10.00</td>
<td>95</td>
<td>1.645</td>
<td>1.400</td>
</tr>
<tr>
<td>LPF_MCO</td>
<td>0.5</td>
<td>0.83</td>
<td>1.67</td>
<td>95</td>
<td>1.645</td>
<td>0.311</td>
</tr>
</tbody>
</table>

Note: The 1.645 is the standard normal variable corresponding to the 95\% upper confidence limit. For the 99\% upper confidence limit, the corresponding value is 2.326.

\*LPF\_MCO = (MCO pressure - 1 atm) / (MCO pressure). At a nominal MCO pressure of 2.0 atm, the LPF\_MCO is 0.5, and at a bounding MCO pressure of 6.0 atm, the LPF\_MCO is 0.83.

\*Nominal\_overall \( M = \frac{MAR\_\text{nominal} \times ARF\_\text{nominal} \times RF\_\text{nominal} \times \text{settling} \times \text{LPF\_MCO} \)}.

\*Bounding\_overall \( M = (\text{EF\_overall}) \times \text{(nominal\_overall \( M \))}.

\( \text{EF\_overall} = \exp \left( NV_{95\%} \left( \sum_i \left[ \frac{\ln(EF_i)}{SNV_i} \right]^2 \right)^{1/2} \right) \).

ARF = airborne release fraction.

EF = error factor (i.e., bounding value divided by nominal value).

ln(EF) = natural log of EF.

LPF = leak path factor.

M = mass of material released.

MAR = material at risk.

MCO = multi-canister overpack.

RF = respirable fraction.

SNV = standard normal variable.

The dose consequences at the remaining receptor sites are calculated in the same manner and are shown in Table A3-23.

#### A3.4.2.4.4 Comparison to Guidelines.

The unmitigated radiological offsite dose for this event is below offsite release limits, and the unmitigated onsite dose does not exceed onsite evaluation guidelines for an anticipated event.
Table A3-23. Dose Calculation Summary for a Hydrogen Deflagration in the Sample Hood.

<table>
<thead>
<tr>
<th>Receptor location (distance, direction)</th>
<th>Duration (hours)</th>
<th>Unmitigated dose* release limits rem (Sv)</th>
<th>Evaluation guideline\textsuperscript{b} (\text{rem (Sv)}) anticipated (\text{mitigated dose rem (Sv)})</th>
</tr>
</thead>
<tbody>
<tr>
<td>Onsite (with building effects) (100 m E)</td>
<td>&lt;1</td>
<td>0.38 &lt;br&gt;(3.8 E-03)</td>
<td>1.0 &lt;br&gt;(1.0 E-02)</td>
</tr>
<tr>
<td>Highway 240\textsuperscript{d} &lt;br&gt;(9,280 m W)</td>
<td>&lt;1</td>
<td>7.8 E-04 &lt;br&gt;(7.8 E-06)</td>
<td>--</td>
</tr>
<tr>
<td>Hanford Site boundary &lt;br&gt;(17,390 m E)</td>
<td>&lt;1</td>
<td>4.3 E-04 &lt;br&gt;(4.3 E-06)</td>
<td>0.5 &lt;br&gt;(5.0 E-03)</td>
</tr>
</tbody>
</table>

\*Fifty-year committed effective dose equivalent.
\textsuperscript{b}Evaluation guideline for onsite (100 m) receptor only.
\textsuperscript{d}Unmitigated frequency for this event is anticipated (>0.01 to \(\leq 0.1\) per year).
\textsuperscript{d}Provided for information only.

A3.4.2.4.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls. No safety-class SSCs are required to prevent hydrogen deflagration DBA accident (WS-L-11) outside of the MCO. Under normal operating conditions, there is no external accumulation of flammable concentrations of hydrogen.

E1. Sample Line Disconnection. The MCO leaves the CVDF with a leak rate less than \(10^{-5}\) cm\(^3\)/s and the sample line leakage also is very low. Under abnormal or accident conditions, safety-significant equipment is required in order to ensure flammable concentrations of hydrogen external to the MCO and CSB systems are precluded. To prevent the bounding external hydrogen deflagration in the sample hood, it is necessary to check the leak rate of the sample line before the MCO port valve is opened. The maximum allowable leak rate is based upon the bounding hydrogen concentration and air flow rate in the sample hood. If the air flow rate is greater than or equal to 5 ft\(^3\)/min in the sample hood and assuming a sample line leak rate up to 40 cm\(^3\)/s, the hydrogen concentration in the hood will not exceed 1%. The 5 ft\(^3\)/min flow rate represents an easily detected minimum, and the 40 cm\(^3\)/s is derived from it (SNF-3328). The following are the specific TSR controls that ensure worker safety:

- TSRs
  - Verify minimum air flow rate for hydrogen gas dilution (5 ft\(^3\)/min) in the hood exhaust for worker safety.

The TSR controls designated to prevent the MCO external hydrogen deflagration are summarized in Table A3-24. Defense-in-depth features also are included in Table A3-24.
<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designator</th>
<th>General function</th>
<th>Safety features and safety classification</th>
<th>NRC ITS category</th>
</tr>
</thead>
<tbody>
<tr>
<td>E1. Sample line disconnection</td>
<td>WS-L-11</td>
<td>Ensure sufficient air flow in the hood to dilute quantities of hydrogen gas leaking from the sample line or connection to below &lt;1% (25% of the lower flammability limit)</td>
<td><strong>Defense in depth:</strong>&lt;br&gt;• Verify minimum air flow rate for hydrogen gas dilution in the hood exhaust&lt;br&gt;• MCO valve operator and sample line (confinement)&lt;br&gt;• Sampling hood exhaust system (hood, ducting, HVAC fan)</td>
<td></td>
</tr>
<tr>
<td>E2. Hydrogen leakage from the MCO inside the MHM</td>
<td>OA-J-06b (in MHM or storage tube)</td>
<td>Mitigate the quantity of release to below onsite consequence guidelines</td>
<td><strong>TSR:</strong>&lt;br&gt;• Verify the proper processing steps and conditions have been carried out at CVDF and K Basins before accepting an MCO at CSB</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td><strong>Defense in depth:</strong>&lt;br&gt;• Limit flow rate of vented gases so that when mixed with air in the portable exhaust duct work, the mixture is not flammable&lt;br&gt;• Limit the contact dose rate on the MHM exhaust system, including the exhaust HEPA filter, to less than 200 mR/h at contact</td>
<td></td>
</tr>
</tbody>
</table>
### E2. Hydrogen Leakage from MCO inside MHM

The hydrogen deflagration in the MHM (OA-J-06b) caused by hydrogen leakage from the MCO occurs during MCO handling. This accident scenario assumes that the MHM contains an MCO with a small leak (10,000 times the criteria). If the MHM is immobilized by a power failure or some other event and the ventilation system is off, the gas mixture in the MHM will become flammable in less than a day (SNF-3328). Combustion of this hydrogen mixture would at most release activity accumulated on the MHM HEPA filters into the environment. Using the bounding release factor for deflagrations on HEPA filters, the onsite dose guidelines are met if the HEPA filter loading is less than 20 g of SNF. Because the corresponding exposure rate near the filter exceeds 600 mR/h, ALARA considerations will ensure the filter loading is well below 20 g. The following specific control that prevent this event:

- **TSR**

  - Verify the proper processing steps and conditions have been carried out at CVDF and K Basins before accepting an MCO at CSB.

### E3. Hydrogen Leakage from MCO inside Cask

The cask–MCO is dropped while being removed from the transportation trailer. The cask is not damaged by this drop, but the MCO may be damaged if the cask falls more than 5 ft. It is assumed that the MCO is breached and loses pressure. The gas leaving the MCO contains considerable hydrogen, so a flammable mixture of

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**Table A3-24. Summary of Safety Features Required to Mitigate the Consequences of a Multi-Canister Overpack External Hydrogen Deflagration. (2 sheets)**

<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designator*</th>
<th>General function</th>
<th>Safety features and safety classificationb</th>
<th>NRC ITS categoryb</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Note: Defense-in-depth features and safety-significant defense-in-depth features are not designated as NRC ITS.


**SSCs** are classified per their function in mitigating and/or preventing specific accidents. **SSCs** may have other classifications based on their functions in other events.

CSB = Canister Storage Building.
CVDF = Cold Vacuum Drying Facility.
HEPA = high-efficiency particulate air (filter).
HVAC = heating, ventilation, and air conditioning.
ITS = important to safety.
MCO = multi-canister overpack.
MHM = multi-canister overpack handling machine.
NA = not applicable to ITS category classification.
NRC = U.S. Nuclear Regulatory Commission.
SSC = structure, system, and component.
SNF = spent nuclear fuel.
TSR = technical safety requirement.
hydrogen, helium, and air forms in the space between the outside of the MCO and the inside of
the cask. This flammable mixture ignites, and the resulting pressure increase may breach the cask.

Calculations have been performed to determine the cask internal pressure following a
combustion event within the cask (SNF-3328). The resulting pressure (about 280 lb/in² gauge)
exceeds the design pressure (150 lb/in² gauge) of the cask. According to WMTS-ECAL-010,
Maximum Pressure Load Determination for MCO Cask, the cask-MCO could sustain a pressure
load up to 345 lb/in² gauge without losing confinement (WMTS-ECAL-010). Therefore, damage
to the cask from combustion is not expected. No additional safety features and controls are
needed to prevent or mitigate this event.

A3.4.2.5 Thermal Runaway Reactions inside the Multi-Canister Overpack. A thermal
runaway reaction is only possible in an MCO containing extremely high temperature fuel and
excessive amounts of water or oxygen. With increasing fuel temperature, the chemical reaction
rates increase and produce more gases (primarily hydrogen) and heat, which increases the
pressure inside the MCO. If pressure inside the MCO continues to increase to the point that the
MCO pressure boundary is challenged, the MCO could fail and release radioactive particulate and
hydrogen gas into the surrounding environment.

The calculations summarized in this subsection demonstrate that a thermal runaway fuel
reaction accident is not credible at the CSB provided the MCOs satisfy the dryness tests
(i.e., < 200 g water) (HNF-1851 [Rev. 1]) at the CVDF and if the aluminum hydroxide thermal
decomposition data based on an initial quantity of 9.47 kg (HNF-SD-SNF-TI-015) and
decomposition rate (SNF-3328) are maintained.

A3.4.2.5.1 Scenario Development. Two primary chemical reactions could lead to a
thermal runaway event in an MCO at the CSB:

T1 — Reaction of water with uranium fuel and uranium hydride (UH₃).
T2 — Reaction of oxygen with uranium fuel and uranium hydride.

T1 — Reaction of Water with Uranium Fuel and Uranium Hydride. The design
pressure of an MCO is 150 lb/in² gauge (HNF-SD-SNF-DR-003) before the cover cap is welded
in place. According to HNF-3312, MCO Monitoring Activity Description, and HNF-3354, MCO
Monitoring Issue Closure Package, a number (around six) of MCOs will be sampled and
monitored after arrival at the CSB. The MCOs reserved for sampling could have time to
pressurize before the MCO cover cap is welded in place. Before welding, an MCO with
mechanical closure is stable to an internal loading of 340 lb/in² (HNF-SD-SNF-SARR-005).
After welding rated pressure is 450 lb/in².

Each MCO is pressurized to approximately 1.5 atm with helium before leaving the CVDF
(SNF-2356). The average temperature of this helium is conservatively assumed to be 25 °C
because the MCO skin is cooled to 25 °C or lower at the CVDF before being shipped to the CSB
(the actual gas and fuel temperatures will be higher). Using the ideal gas law, the initial helium
inventory in an MCO is estimated to be about 33 gmoles (SNF-3328). The number of moles of
gas required to increase the MCO pressure to 150 lb/in² gauge and to 450 lb/in² gauge can be
computed from the ideal gas law. For an MCO gas temperature of 150 °C, which is greater than
the bounding value of 125 °C (HNF-SD-SNF-TI-015), 173 moles of gas would have to be
present in the MCO to achieve a pressure of 150 lb/in² gauge.

At a rate dependent on temperature and steam pressure, water (liquid or vapor) reacts with
uranium to form uranium dioxide particulate and hydrogen gas, liberating heat in the process.
Water also reacts with uranium hydride (UH₃) to form uranium dioxide and hydrogen, again
liberating heat during the reaction. About 33 moles of helium are contained in an MCO at the
time of arrival at the CSB. An additional 140 moles of additional gases would need to be created
to reach 150 lb/in² gauge for an MCO gas temperature of 150 °C.

The amount of water needed to generate this additional quantity of hydrogen depends on
whether uranium or uranium hydride is reacting. Table A3-25 summarizes the total free water
required to react in order for the MCO to reach pressures of 150 lb/in² and 450 lb/in² gauge.

Table A3-25. Water Mass Required to Pressurize Multi-Canister
Overpack by Chemical Reaction.

<table>
<thead>
<tr>
<th>MCO pressure</th>
<th>150 lb/in² gauge</th>
<th>450 lb/in² gauge</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gas temperature</td>
<td>125 °C</td>
<td>125 °C</td>
</tr>
<tr>
<td>Combined uranium metal and hydride</td>
<td>2.15 kg</td>
<td>8.20 kg</td>
</tr>
<tr>
<td>reaction with water (weighted)*</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

*Weighted combination based on bounding uranium hydride inventory and reaction surface area
from HNF-SD-SNF-TI-015, 1998, Spent Nuclear Fuel Project Technical Databook, Rev. 6, Fluor Daniel
Hanford, Incorporated, Richland, Washington.

After successful processing at the CVDF, the bounding inventory of free water in an MCO
received at the CSB is 200 g. The bounding MCO is assumed to be dried at the CVDF, with less
than 200 g of free water remaining in cracks after the dryness tests at the CVDF (HNF-1851
[Rev. 1], HNF-SD-SNF-TI-015). In addition to the 200 g of free water, there is a bounding value
of about 1.19 kg of water in the uranium hydrate that is part of the uranium oxide particulate
matter (HNF-1523 [Rev. 1], HNF-SD-SNF-TI-015) for an MCO with two scrap baskets and
three fuel baskets. However, according to HNF-SD-SNF-CN-023, Thermal Analysis of Cold
Vacuum Drying of Spent Nuclear Fuel, this water would not be initially available for hydrogen
gas-producing reactions, and some of the hydrate water is expected to be removed at the CVDF.
The bounding water content MCO is one with two scrap baskets and three fuel baskets with fuel from aluminum canisters at the K Basins. For this MCO, up to 3.32 kg of water is contributed by both the bounding quantity of aluminum hydroxide, Al(OH)$_3$, on the fuel cladding and by an additional 0.13 kg of water in aluminum and iron hydrates in the canister sludge (HNF-1523 [Rev. 1], HNF-SD-SNF-TI-015). This bound water, like the uranium hydrate water, would not initially be available for reactions. Very little of the hydroxide water is expected to be freed from the thermal decomposition of the hydroxide based on previous and current data (HNF-1523 [Rev. 0 and Rev. 1]) at fuel temperatures below 150 °C. The data set provided in SNF-3328 and HNF-1523 (Rev. 0), is used because it is more conservative at lower temperatures (<150 °C) than the recent data (HNF-1523 [Rev. 1]). Both data sets are similar at high temperatures around 300 °C.

The expected bounding sources of water in an MCO on arrival at the CSB and the availability of the water for reactions are shown in Table A3-26. If the MCO temperatures reach very high values (approximately 200°C), the thermal decomposition of aluminum hydroxide, along with hydrate decomposition and initial free water, could supply about 1.91 kg of water. This total is based on 15% thermal decomposition of the water in aluminum hydroxide and the water in Al+Fe hydrate, 100% thermal decomposition of the uranium hydrate, and the 0.2 kg of free water.

<table>
<thead>
<tr>
<th>Source of water</th>
<th>Total possible water mass$^a$</th>
<th>Availability of water for reactions at given fuel temperatures (thermal decomposition, percent of total water mass)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>$&lt; 100°C$</td>
</tr>
<tr>
<td>Free water in cracks</td>
<td>0.20 kg</td>
<td>0.20 kg</td>
</tr>
<tr>
<td>Water in uranium hydrate</td>
<td>1.19 kg</td>
<td>0.60 kg (50%)</td>
</tr>
<tr>
<td>Water in aluminum hydroxide</td>
<td>3.32 kg</td>
<td>0.20 kg (6%)$^b$</td>
</tr>
<tr>
<td>Water in Al+Fe hydrates</td>
<td>0.13 kg</td>
<td>0.0 kg</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total</td>
<td>4.84 kg</td>
<td>1.00 kg</td>
</tr>
</tbody>
</table>


The water required (Table A3-25) is combined with the water available (Table A3-26) to determine the additional free water needed to pressurize the MCO to 150 lb/in² gauge and 450 lb/in² gauge. It is assumed in Table A3-27 that the fuel temperatures are not more than 50 °C higher than the gas temperatures, which has been shown to be true for most conditions (HNF-SD-SNF-CN-023).

### Table A3-27. Additional Water Mass Needed to Pressurize Multi-Canister Overpack.

<table>
<thead>
<tr>
<th>Water balance description</th>
<th>Gas temperature 125 °C</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>150 lb/in² gauge</td>
</tr>
<tr>
<td>Water required to reach rated pressure</td>
<td>2.15 kg</td>
</tr>
<tr>
<td>Free water available (Table A3-25)</td>
<td>1.91 kg</td>
</tr>
<tr>
<td>Additional water needed to reach 150 lb/in² gauge (water shortage at the CSB)</td>
<td>0.24 kg</td>
</tr>
</tbody>
</table>

CSB = Canister Storage Building.

In summary, Table A3-27 shows that for bounding conditions, at least 6.29 kg of additional free water would need to be available at the CSB for the MCO pressure to reach 450 lb/in² gauge. However, the MCO design pressure is 150 lb/in² gauge before welding, therefore the additional water margin for the unwelded MCO is much smaller, as shown in Table A3-27. In fact, only 0.24 kg of additional water is needed to pressurize the MCO to 150 lb/in² gauge at the CSB under severe conditions. However, the unwelded MCO is not expected to leak at 150 lb/in² gauge.

HNF-SD-SNF-SARR-005 states that based on the codes used in the MCO design, the mechanically closed MCO (before welding at the CSB) would not suffer any damage at internal pressures up to 340 lb/in² gauge. At some pressure in excess of 340 lb/in² gauge, the thread root of the collar would begin to yield, and at some pressure very much greater than 340 lb/in gauge, leakage around the Helicoflex seal would result. An MCO has been built and tested to an internal pressure of 562.5 lb/in² gauge to match the test requirement to be placed on MCO procurement. In that test, no leakage and no permanent damage were observed.

Because the additional water margin is small for the 150 lb/in² gauge pressure and the calculations in Table A3-27 are only approximate, the HANSF code was used to simulate a very hot MCO in the sampling/weld station under severe conditions. The software verification and validation documentation for version 1.2 of the HANSF code may be found in FAI/98-40, *Hanford Spent Nuclear Fuel Safety Analysis Model HANSF 1.2: User's Manual*. HNF-SD-SNF-CN-023 also documents that the HANSF code has been benchmarked for code version 1.2 using the benchmark data in FAI/98-40. The key input parameters are given in Appendix D of SNF-3328 based on HNF-SD-SNF-TI-015. The newer HANSF version 1.3.2, described in SNF-3650, *HANSF 1.3.2: User's Manual*, is compared with HANSF version 1.2 in SNF-5226, *Comparison Cases Simulated with HANSF 1.3.2 that Supplement the Thermal...*
Analyses Documented in HNF-SD-SNF-CN-023. The comparison shows that the results from HANSF version 1.2 are more conservative than the results from HANSF version 1.3.2 for the cases in this analysis.

To maximize the results and minimize the number of simulations, a hypothetical MCO that had the decay heat of an MCO with one scrap basket and the water content of an MCO with two scrap baskets was simulated. Although five fuel baskets may have more decay power (~35 W) than four fuel baskets and one scrap basket, the scrap basket has more reactive surface area and will contribute more chemical heat when oxidants are present (SNF-3328).

In the first case, a bounding MCO is left in the sampling/weld station for at least 40 days without active cooling (e.g., because of MHM failure). The calculated fuel temperatures are consistent with the 132 °C boundary (MCO wall) temperature. The simulations were started with fuel temperatures at 125 °C and calculated the temperatures for 2 days. Water was assumed to be 2% (saturated steam at 25 °C) of the 1.5 atm of helium injected at the CVDF, or about 9 g of water mass. In this simulation, steady-state temperatures for this case are attained in less than a day (SNF-3328).

The highest fuel temperature for this scenario occurs on the inner fuel elements (six total) nearest the center post of the MCO and has a value of about 145 °C. The highest temperature for the scrap fuel is about 140 °C, with a steady-state value of 135 °C, which is cooler than the hotter fuel elements because the copper fins on the scrap basket effectively conduct heat toward the MCO. This simulation uses the maximum decay power for fuel from aluminum canisters. According to HNF-3035, MCO Gas Composition for Low Reactive Surface Areas, the bounding decay power for fuel from aluminum canisters is 528 W per five fuel baskets.

The MCO gas temperature reaches about 140 °C in the fuel baskets and about 132 °C in the scrap basket. The initial gas temperature is 25 °C when helium is injected into the MCO at a pressure of about 1.5 atm at the CVDF. This cool gas heats up very quickly in the simulation and causes a rapid pressure increase. It is anticipated that the MCO wall temperatures will be well below the 132 °C limit.

In another case, the final results of the previous case were used for all initial conditions. This simulation adds 0.52 kg of water, which is about 15% of the water contained in aluminum hydroxide and the aluminum and iron hydrates in the canister sludge (HNF-SD-SNF-TI-015). ALCOA data (SNF-3328) indicates that as much as 15% of the hydroxide water (or about 5% of total hydroxide mass) can be freed by thermal decomposition for temperatures up to 200 °C and about 57% for temperatures up to 300 °C. Since the maximum fuel temperature in the previous case was 145 °C, indicating that less than 15% of the water in aluminum hydroxide is expected to be freed. Because thermal decomposition of aluminum hydroxide is not included in the HANSF code, additional water vapor (steam) was added as a source to simulate its effect on temperatures and pressure. To be conservative, 15% or 0.52 kg of hydroxide water was added to this case over a short 10,000-second interval. The Alcoa data set used in SNF-3328 is more conservative at lower temperatures than recent data (HNF-1523 [Rev. 1]). One reason for the lower
decomposition rate found in the data set used in SNF-3328 may be that thermal equilibrium was not achieved.

In the simulation, the freed hydroxide water was added only to the fuel baskets, which is more conservative than spreading the water into all of the baskets. The added hydroxide water causes the maximum fuel temperature to increase to about 155 °C in less than three hours. The fuel and gas temperatures decrease after 3 hours because no more water is available to continue the chemical reactions. In less than 2 days, the maximum fuel temperature reaches a new steady-state value of 145 °C, which is lower than the temperature before the additional water was added. The additional water created hydrogen gas, which has a high thermal conductivity, so heat is removed faster from the MCO. The MCO pressure rises to 10.0 atm (132 lb/in² gauge) (SNF-3328). This is below the MCO design pressure of 11.2 atm (150 lb/in² gauge). The pre-welded (i.e., mechanically sealed) MCO can withstand pressure up to 340 lb/in² (HNF-SD-SNF-SARR-005). The maximum fuel temperature does not rise above 155 °C, and the gas temperature does not rise above 140 °C, indicating that the helium and hydrogen provide good thermal conductivity, thereby keeping the temperatures stable in the MCO at the CSB.

Conclusions and Conservatisms for Water Reaction Cases. MCO temperatures have been shown to be stable even under very severe external thermal conditions, and the maximum gas pressure has been shown to stay below 150 lb/in² gauge, which is far below the 450 lb/in² gauge MCO design pressure after the cover cap is welded in place. The main conservatisms and/or margins over bounding parameter values in the computer simulations are itemized in the following list:

- No hydrate water is removed at the CVDF, so all hydrate water is available for thermal decomposition at the CSB.
- No hydride mass is consumed at the CVDF and a hydride reaction rate multiplier of 12 is used for all simulations at all times (this is documented in HNF-SD-SNF-TI-015).
- MCO wall temperature is 132 °C because of being located in the sampling/weld station pit without active cooling for 40 days (this is documented in CSB-HV-0014).
- Steam (water vapor) mass of 9 g is added to the MCO to account for the CVDF helium supply possibly being contaminated with 2% steam.
- Fifteen percent of aluminum hydroxide water is released at a lower temperature (<150 °C) instead of at a higher temperature of 200 °C (this is documented in HNF-1523 [Rev. 0]), and this water is added only to the fuel baskets instead of being spread among the fuel and scrap baskets.
- No hydrogen gettering takes place, which maximizes the gas pressure; if hydrogen gettering were allowed to take place, the hydrogen fraction in the MCO would get
very small and reduce the pressure by a large amount. Hydrogen gettering is expected to occur after all of the free water has been depleted, which could lower the MCO pressure by as much as 80%.

T2 — Reaction of Oxygen with Uranium Fuel and Uranium Hydride. Another way to generate heat in an MCO is for oxygen (or air) to enter as the result of an accident event or to be created by radiolysis. Section A3.4.2.3 examines the long-term effects of radiolysis and flammability issues.

Any oxygen that enters the MCO will react with uranium hydride and uranium to liberate heat, depending on the temperature (HNF-SD-SNF-TI-015). Reactions with oxygen liberate heat and increase the fuel and gas temperatures so that additional water would be available from the decomposition of the uranium hydrate and aluminum hydroxide, which would increase the pressure and temperature in the MCO. The reactions with oxygen are followed by the reactions with water after the oxygen is consumed. Because all of these reactions are coupled and interrelated, the HANSF code (FAI/98-40) was used to simulate air ingress.

In one case, an MCO is charged with pure oxygen instead of helium during gas sampling at the CSB sampling/weld station (i.e., helium bottles accidentally filled with oxygen or oxygen bottles accidentally used in place of helium). In obtaining a gas sample from one of the monitored MCOs, the helium pressure in the MCO is accidentally reduced to 1.0 atm. Hence, when oxygen, instead of helium, is accidentally injected into the MCO to get the pressure back up to 1.5 atm, 0.5 atm (about 33%) of the total pressure is due to oxygen. Heat removal is assumed to be limited in the sampling/weld station by a cooling failure of the shield, such that the air temperature and MCO wall temperature are both 132 °C, the maximum possible with complete cooling failure (CSB-HV-0014). This maximum steady-state temperature is only reached after 40 days in the sampling/weld station with no active cooling. All fuel temperatures are conservatively assumed to be at 153 °C in the HANSF simulation.

A very high fuel temperature was assumed to demonstrate margin. If oxygen reactions starting at high temperatures do not cause thermal excursion, then oxygen reactions starting at lower temperatures will not cause thermal excursion. This scenario is expected to bound all air entry cases and cases with helium bottles contaminated with air or oxygen because 100% oxygen is postulated to be injected and air has an oxygen content of only 21%. Also, no natural circulation with air ingress is expected at the CSB because natural circulation requires two openings, one for air entry and one for gas exit, and two openings are not available in the MCO at the CSB.

Therefore, oxygen, instead of helium, is assumed to be injected into the MCO at the gas sampling station. The innermost fine scrap fuel has a thermal increase up to about 440 °C before the oxygen is depleted in the scrap basket and cools off. This temperature increase occurs because the fine scrap fuel has a high surface-area-to-volume ratio and is initially hot enough (153 °C) to rapidly oxidize the uranium hydride in the scrap fuel and increase the fine scrap temperature dramatically. The oxygen-hydride reaction at this high temperature rapidly consumes
the oxygen and is depleted within an hour. Because the scrap basket is dry, except for the thermal
decomposition of the hydrate mass, there is not enough water available to continue the chemical
reactions. If aluminum hydroxide were included, only a small amount of water would be freed by
thermal decomposition because the elevated temperature lasts for less than an hour and is
restricted to only the innermost fine scrap in the fine scrap portion of the scrap basket
(SNF-3328).

The maximum MCO pressure of about 6.2 atm in 1 day is far below the MCO design
pressure of 150 lb/in² gauge before the cover cap is welded in place. The peak pressure is lower
than the previous cases (with water) because oxygen reactions do not produce as much gas as
water reactions. Also, the temperatures in this case are not as high, except for the sharp peak, as
in the previous cases, so the hydrates do not completely decompose for this case, which means
less water reacts in this case than in the previous two cases.

Conclusions and Conservatisms for Oxygen Reaction Case. Temperatures remain stable in
an MCO even under very severe external thermal conditions. The following conservatisms and/or
margins over bounding parameter values in the computer simulations ensure that the MCO will
not exceed its design pressure limit.

- No hydrate water is removed at the CVDF, so all hydrate water is available for
  thermal decomposition at the CSB.

- No hydride mass is consumed at the CVDF and the hydride reaction rate multiplier is
  kept at 12 (HNF-SD-SNF-TI-015) for all times, which keeps the hydride-oxygen
  reaction going. In reality, much of the hydride would be consumed at the CVDF and
  not be available for reaction at the CSB.

- The wall temperature of the MCO is 132 °C because of failure of the MHM and of
  the cooling shield in the sampling/weld station for 40 days.

A3.4.2.5.2 Source Term Analysis. Because there is no release expected even under
severe conditions, no source term was estimated.

A3.4.2.5.3 Consequence Analysis. Because there is no release expected, even under
severe conditions, the inhalation dose consequences are zero.

A3.4.2.5.4 Comparison to Guidelines. Because the dose consequences are zero for
thermal runaway reaction events at the CSB, all dose guidelines are met. These results assume
that the MCOs pass the dryness tests at the CVDF and that the tests are accurate in ensuring that
less than 200 g of free water is present in the MCO after leaving the CVDF. The results depend
on the amount of hydride in the MCO and the amount of aluminum hydroxide, as well as its
thermal decomposition rate as a function of temperature. If the MCO design pressure were
lowered to 150 lb/in² gauge, there would still be no dose consequences, but the margin of safety
would be significantly reduced.
The equivalent cases involving MCOs with two scrap baskets, analyzed and presented here, maximize the amount of water and hydrides in the MCO. These cases are expected to bound the thermal and high-pressure estimates and associated consequences when compared with events involving an MCO with one scrap basket or a MCO with no scrap baskets.

A3.4.2.5.5 Summary of Safety Structures, Systems, and Components and Technical Safety Requirement Controls. All of the hazards related to fuel reactions that could cause a release are shown in Table A3-28. Detailed analyses in Section A3.4.2.5.1 showed that there was not enough heat, air, or water in the MCO for any of these initiators to cause thermal runaway events. Therefore, no safety-class or safety-significant SSCs are required to prevent the consequences of this accident.

T1. Thermal Runaway Reaction Due to Fuel Reaction with Water. The fuel reactions with water hazards were identified with zero-release consequences in the CSB hazard analysis because of insufficient free water (HNF-SD-SNF-HIE-001). The following TSR protects the assumptions used in the analysis:

- TSR
  - Verify the proper processing steps and conditions have been carried out at CVDF and K Basins before accepting an MCO at CSB.

T2. Thermal Runaway Reaction Due to Fuel Reaction with Oxygen or Contaminated Gas. Temperatures are shown to remain stable in an MCO even when the MCO is presumed to be inerted with pure oxygen or an air ingress occurs.

No SSCs are required to prevent the MCO thermal runaway accident. The TSR and defense-in-depth features are summarized for each specific accident in Table A3-28.
Table A3-28. Summary of Safety Features Required to Prevent a Multi-Canister Overpack Thermal Runaway.

<table>
<thead>
<tr>
<th>Accident</th>
<th>Checklist designator</th>
<th>General function</th>
<th>Safety features and safety classification</th>
<th>NRC ITS category</th>
</tr>
</thead>
<tbody>
<tr>
<td>T1. Thermal runaway reaction — fuel reaction with water</td>
<td>SA-J-10b, OA-J-10b, WS-J-10b</td>
<td>None at the CSB required for safety</td>
<td>General assumption is that the MCO is received within specifications.</td>
<td></td>
</tr>
<tr>
<td>T2. Thermal runaway reaction — use of contaminated or wrong gas for inerting</td>
<td>WS-H-06b</td>
<td>None at the CSB required for safety</td>
<td>Defense in depth:</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Verify the proper processing steps and conditions have been carried out at CVDF and K Basins before accepting an MCO at CSB.</td>
<td></td>
</tr>
</tbody>
</table>

Note: Defense-in-depth features and safety-significant defense-in-depth features are not designated as NRC ITS.


SSCs are classified per their function in mitigating or preventing specific accidents. SSCs may have other classifications based on their functions in other events.

CSB = Canister Storage Building.
CVDF = Cold Vacuum Drying Facility.
ITS = important to safety.
MCO = multi-canister overpack.
NA = not applicable to ITS category classification.
NRC = U.S. Nuclear Regulatory Commission.

A3.4.2.6 Violation of Design Temperature Criteria. Heat is produced in an MCO from energy released by radioactive decay and by chemical reactions occurring between the fuel and water or gases. MCOs and the CSB have been designed to provide for sufficient heat transfer from the MCO so that unacceptably high temperatures will not be reached during normal handling and storage of the MCO at the CSB. This analysis investigates situations in which a reduction in normal heat conduction causes overheating of an MCO and surrounding structures. Preventive measures are required to preclude this overheating because it could compromise the safety function of safety components. Required controls to prevent the potential consequences of each overheating condition are summarized in Section A3.4.2.6.5. Additional information about the violation of design temperature criteria accidents can be found in SNF-3328.

A3.4.2.6.1 Scenario Development. Radioactive decay reactions and chemical reactions within the MCO generate heat. The situation in which heat transfer from the MCO to the outside environment is disrupted enough to lead to violation of the temperature criteria was identified in...
the CSB hazard analysis (HNF-SD-SNF-HIE-001). In this case, the natural convective cooling airflow to the vault is disrupted. This bounds effects due to convective heat transfer at other locations in the CSB. The hazard is exceeding the design temperature for the concrete vault structure. See HNF-SD-SNF-SARR-005, Chapter 4.0, for thermal design limits of the MCO.

If the passive cooling air flow of the CSB vault is significantly reduced, either the MCO wall temperature or the CSB vault concrete temperature could exceed its respective design temperature. The natural convection of air through the vault will be reduced for blockage (full or partial) of vault intake and/or exhaust stacks. Significant blockage of the flow occurring from causes including debris trapped in the inlet or frost formation over the inlet is very unlikely. Without the vault convective air cooling, temperatures in the vault would rise above the design temperature criteria for the vault walls and operating deck, thereby reducing the strength of the concrete and compromising their safety functions. Loss of the structural integrity of the vault concrete could make it more susceptible to failure during a seismic event.

ACI-349, Code Requirements for Nuclear Safety Related Concrete Structures, Appendix A.4, provides limitations for concrete temperatures during normal operation or any long-term period. The temperatures shall not exceed 66 °C (150 °F), except for local areas (e.g., around penetrations), which are allowed to have increased temperatures not to exceed 93 °C (200 °F). The concrete surface temperature shall not exceed 177 °C (350 °F) for an accident duration or any short-term period.

Bounding calculations for the effect of loss or partial loss of vault convective cooling flow on the MCO wall and the vault concrete temperatures have been performed and are documented in CSB-HV-0003, Spent Nuclear Fuel Vault Loss of Cooling Analysis. These calculations assume that the vault is full of MCOs with a corresponding total heat load of 191.6 kW (HNF-SD-SNF-TI-015). The outside air temperature available to cool the vault is assumed to be 46 °C (115 °F), and the surrounding ground temperature is assumed to be 16 °C (60 °F). With these conditions, the maximum initial temperature of the MCO wall (no blockage of vault air flow) is about 122 °C (252 °F), and the maximum vault concrete ceiling surface temperature is 56 °C (132 °F). CSB-HV-0003 calculations show that for an intake stack cross-sectional area reduction of 25% of normal flow (i.e., is 75% blocked), the MCO wall temperature reaches 131 °C (268 °F), and the operating deck temperature reaches 66 °C (151 °F) in 72 hours. For a complete vault inlet stack blockage, the MCO wall temperature reaches 132 °C (270 °F) in about 15 hours with the temperature continuing to rise at a rate of about one degree per hour. For 50% or less of the inlet area blocked, the vault air flow rate and the MCO and vault concrete temperatures would be nearly equal to those expected for zero blockage.

The bottom of the inlet area of the vault intake structure is more than 62 ft above the surrounding grade. A vertical grating surrounds the opening to the intake stack on four sides. The grating consists of a heavy gauge (0.13-in. wire diameter) interwoven stainless steel screen with 0.75-in. square openings supported by more widely spaced steel cross members. The overall dimensions of this grating are about 15 ft by 17 ft on two sides and 15 ft by 18 ft on two sides, giving a total flow area for all four vertical inlet grates of more than 1,000 ft². Inlet air flows
through this grating and down the inlet stack to a second horizontal grate inside the inlet stack. The horizontal grating is composed of a metal lattice with perpendicular members spaced 6 in. apart. The intake stack horizontal cross-section is rectangular with minimum interior dimensions of about 16 ft by 11 ft and has a flow area of more than 170 ft². Blowing debris (e.g., tumbleweeds or garbage) may occasionally be lifted to the height of the upper vertical grating and become lodged but by design, significant (>50% blockage) accumulation of debris on either grating is not considered possible for the following reasons.

- The intake outer grating is open on all sides and the opening is quite large, it is not probable that a significant portion of the intake could become blocked.
- The openings in the vertical inlet grate are significantly smaller than those of the horizontal grate, it is not considered possible for significant debris to accumulate on the horizontal grate.
- The vertical inlet grating has a total flow area that is more than 6 times that of the horizontal cross-section of the inlet, about 92% of the vertical grating area must become fully blocked before more than 50% of the inlet cross-section area will be blocked.
- Both the intake and exhaust are significantly above grade.

In summary, the bounding scenario for this accident category describes conditions in which CSB safety classified concrete structures could exceed their design temperatures because of a lack of cooling. The unmitigated scenario is brought to a stable state by restoring cooling to bring the vault to a stable thermal condition. If the MCOs are in the vault of the CSB, any debris is removed to restore air flow to the air intake and exhaust structures. Once a stable thermal condition is attained, the MCO is handled in accordance with recovery procedures or the vault is returned to operational status.

**A3.4.2.6.2 Source Term Analysis.** The scenario described above could potentially damage safety-class structures (in this case, the vault) and result in compromise of their safety function. This accident also leads to the potential loss of structural capability for the concrete vault structures. Controls as defined in DOE Order 6430.1A are required to prevent conditions that would lead to the damage of safety-class equipment or structures that would impede their ability to perform their safety function. Because these accidents are prevented, no source terms are developed.

**A3.4.2.6.3 Consequence Analysis.** The potential unmitigated impact is a reduction in the structural integrity of the concrete. However, the maximum expected vault temperatures are not sufficient to cause loss of structural capacity.

Temperature increases in the vault over a long period of time result in gradual damage to the vault and operating deck concrete. Damage to the concrete would slowly continue once it
began. A structurally impaired operating deck may not survive the DBE. The collapse or
significant failure of the deck could potentially lead to the breach of MCOs stored in the vault.
Ample time would be available to identify and correct any loss of air flow condition in the vault.
Because the outer vertical inlet screens are so large and are located so far above the existing
grade, it is not considered credible for sufficient blockage of the inlet to occur and result in a
temperature violation. These temperature criteria violation scenarios will be prevented so that no
safety-class structures or equipment will be damaged.

Calculation CSB-S-0043C, *HCSA Deck Design — Confirmation*, evaluates the thermal
effects on a concrete structure of an MCO left in the sampling/weld pit for 40 days. Using the
thermal gradients shown in CSB-HV-0014, *Long Term MCO Temperature Without Cooling in
the Sampling Station*, the deck and the pit meet all design criteria (worst-case demand—capacity
ratio is 0.99 [vertical reinforcing on inside face of the tubular portion of the pit]). The calculation
considered concrete strength degradation from thermal exposure by reducing the concrete
strength from 4,000 lb/in² to 3,000 lb/in². This approach is conservative and has been used in
other CSB structural calculations to account for concrete degradation.

**A3.4.2.6.4 Comparison to Guidelines.** Safety-class SSCs are identified to keep vault
temperatures from exceeding the maximum expected values, thus no comparisons to dose
consequence guidelines are made to identify additional SSCs.

**A3.4.2.6.5 Summary of Safety Structures, Systems, and Components and Technical
Safety Requirement Controls**

**VI. Loss of Vault Cooling.** The safety-class structural components that are required to
ensure that passive cooling is sufficient to prevent the MCOs or safety-class facility concrete
structures from being exposed to temperatures in excess of design criteria are summarized in
Table A3-29 and listed below:

- Safety-class SSCs
  - Vault (concrete) — Maintains geometry to provide for passive convective
    cooling air flow past the storage tubes
  - Vault intake structure — Provides inlet for passive cooling air to the vault, is
    above grade, and has screened entrances that provide assurance that the inlet
    will not become significantly obstructed
  - Vault exhaust stack — Provides exit for passive cooling of vault, is above
    grade, and is designed such that it will not become significantly obstructed
  - Storage tubes, tube base assemblies, and carbon steel base slab embeds —
    Provide spacing of MCOs within the vault to ensure that each will be cooled
    by passive airflow.
Table A3-29. Summary of Safety Features Required to Prevent Violation of Design Temperature Criteria Design Basis Accidents.

<table>
<thead>
<tr>
<th>Accident Description</th>
<th>Checklist Designator</th>
<th>General Function</th>
<th>Safety Features and Safety Classification</th>
<th>NRC ITS Category</th>
</tr>
</thead>
<tbody>
<tr>
<td>V1. Loss of vault cooling</td>
<td>VL-B-07, VL-B-10, VL-B-11</td>
<td>Maintain structures and configuration required to maintain adequate passive convective cooling flow through the vault</td>
<td>Safety-class SSCs: - Vault, vault intake structure (including screen), vault exhaust stack, carbon steel basemat embeds, tube base assemblies, and storage tubes</td>
<td>A</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Defense in depth: - Differential temperature monitors are located on outlet stack and give indication of undesired trends - Various screens on the inlet and outlet stack reduce the likelihood of accumulation of debris in the vault</td>
<td></td>
</tr>
</tbody>
</table>

Note: Defense-in-depth features and safety-significant defense-in-depth features are not designated as NRC ITS.


SSCs are classified per their function in mitigating or preventing specific accidents. SSCs may have other classifications based on their functions in other events.

ITS = important to safety.
NRC = U.S. Nuclear Regulatory Commission.
SSC = structure, system, and component.

A3.4.3 Beyond Design Basis Accidents

DOE Order 5480.23 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 BDBAs are not analyzed to the same level of detail as DBAs, but rather simply provide insight into the magnitude of consequences of BDBAs. This insight from BDBA analysis has the potential for identifying additional facility features that could prevent or reduce severe BDBA consequences. Only BDBAs of significance (for internal events and natural phenomena hazards) are identified and assessed for their importance.

This section presents a summary of the key assumptions and results of the BDBA analyses that have been performed for the CSB. As such, the evaluation associated with each BDBA provides insight into the magnitude of the consequences of BDBAs.

Identified CSB BDBAs include the following:

- Failure of MCO confinement because of internal overpressurization
- Hydrogen combustion events
• Collapse of the building’s superstructure

• Failure of multiple MCOs

• Failure of identified safety-class accident prevention and mitigation features, including those that control drops, collisions or impacts.

The DBA analyses in Section A3.4.2 envelop the scenarios and consequences of failure of MCO confinement because of internal overpressurization, and hydrogen combustion events. The analyses demonstrate that these events are limited in their potential hazardous material quantities and energy sources because of limited reactants in the MCO. The analyses assume that the safety basis properties and facility features are in place to ensure the reactants are limited. An MCO received out of these specifications is a BDBA discussed in Section A3.4.3.1.

Collapse of the CSB superstructure could be caused by human-generated external events (e.g., crash of a light airplane) or natural phenomena (e.g., tornado and tornado-generated missiles, high winds, earthquake, or snow or ashfall loading of the roof). DOE-STD-3009-94 specifically excludes evaluation of human-generated events as BDBAs. The consequences of beyond design basis natural phenomena events are generally bounded by a beyond design basis seismic event, which is discussed in Section A3.4.3.2. An exception would be a beyond basis natural phenomena event that leads to flooding of the CSB underground vaults. Flooding of the vaults is discussed in Section A3.4.3.3.

Failure of multiple MCOs (e.g., as a result of a seismic event) is considered in the discussion of common mode initiators in Section A3.4.4. The failure of identified safety class accident prevention and mitigation features (including those that control drops, collisions, or impacts) is discussed in Section A3.4.3.4.

The results of the BDBA evaluations did not identify additional mitigative or preventive barriers beyond those required for the DBAs.

A3.4.3.1 Failure of Multi-Canister Overpack Confinement Because of Internal Overpressurization. As discussed in HNF-SD-SNF-OCD-001, *Spent Nuclear Fuel Project Product Specification*, receipt of MCOs with properties outside of safety basis assumptions is beyond the design basis of the SNF Project. As such, this is a BDBA occurrence that could lead to consequences at the CSB beyond those described in the DBAs in Section A3.4.2. Key properties relied on to ensure that the as-received content and condition of the mechanically sealed MCO (150 lb/in²) are as required for safety basis properties include the following values (see Section A3.4.2):

1. Maximum particulate mass of aluminum hydroxide in an MCO delivered to the CSB less than or equal to 9.47 kg
2. Maximum mass of uranium hydrates in an MCO delivered to the CSB less than or equal to 10.8 kg

3. Maximum of two scrap baskets in an MCO delivered to the CSB

4. Maximum removable, not strongly adherent, UO₂ particulate in an MCO at the CSB is 34 kg

5. Maximum free water in an MCO delivered to the CSB less than 200 g

6. Leak rate of internal gas from MCO less than 10⁻⁵ standard cm³/s

7. Maximum mass of uranium hydride (for heat and hydrogen generation) of 5.13 kg on receipt at CSB and 9.08 kg after 40 years storage at CSB

8. Maximum of 6,339 kg of uranium per MCO (five Mark IV fuel baskets)

9. Minimum MCO inert gas pressure on receipt at CSB of 1.5 atm at 25 °C.

An overpressurization is the most significant consequence of excess free and bound water inside the MCO. Such an occurrence is related to exceeding limits 1, 2, 3, and 5, individually or in combination. Because the uranium reaction with water is exothermic, heat released by the reaction could raise the temperature of the fuel to the point of a thermal runaway if the MCO contained excess amounts of water available for reaction. Additional scrap baskets mean more fuel surface is available for reaction, leading to potentially more severe fuel oxide exothermic reactions.

Exceeding limits 1, 2, and 5, individually or in combination, could also result in internal hydrogen deflagrations because the increased water available for radiolysis allows increased production of hydrogen gas. Exceeding limit 6 by increasing the leak rate could increase the probability of occurrence and consequence of external hydrogen deflagration events.

The integrated facility dose resulting from an out-of-specification MCO caused by overpressurization can be estimated from the DBA for a gaseous release from the MCO, Section A3.4.2.2. The MAR from the overpressurization would be three to four times the MAR for the gaseous release DBA because the generation of the high pressure required to fail the MCO also creates additional amounts of UO₂ during the reaction. For example, 450 lb/in² of H₂ pressure requires the reaction of water with approximately 90 kg of UO₂ compared with the 35 kg of UO₂ available for the gaseous release DBA.

An increase of three to four times in the MAR would essentially result in an increase of three to four times in the radiological dose consequences at both the onsite and offsite receptors. However, three to four times the radiological dose consequences calculated for the gaseous release DBA still represents a relatively small dose to receptors of interest. Finally, although not
credited for the DBA, the MHM extract system and the building exhaust, and their associated
HEPA filters, provide additional mitigation of some portion of the release.

A3.4.3.2 Beyond Design Basis Seismic Event. Beyond design basis forces from seismic events
could affect the performance of systems and structures in the CSB during MCO handling and
placement operations and during interim storage. The impact of a beyond DBE on structures is
discussed in Section A3.4.4.1. The CSB has been designed to withstand seismic forces greater
than the Hanford Site DBE with the functions of the structure, including spent fuel cooling,
remaining intact. As discussed in Section A3.4.4.1, seismically induced failures of the
sampling/weld systems at CSB would have only significant onsite consequences.

When the MCOs are in their storage tubes, the facility structures, storage tube assemblies,
and MCO are passive structural barriers that are all designed for the DBE. In the beyond DBE
event, failures of above-grade structures (e.g., the operating area shelter, support area building,
and intake and exhaust stacks) would not significantly affect the structural integrity of the
below-grade storage tubes and MCOs. If the flow of air through the vault were impeded,
affecting the cooling capability of the vault, the MCO skin and fuel temperature criteria could be
exceeded over a long period of time. Significant blockage, more than 50% of the intake structure
cross-sectional area, could result in thermal conditions approaching the thermal criteria in a few
days, which is sufficient time for recovery actions. Failures of the above-grade structures with
below-grade structures intact would allow the below-grade structures to continue to provide a
reasonably stable and safe condition for the SNF with sufficient time to recover adequate passive
cooling capability before the MCO thermal criteria are exceeded. Thus, source terms and dose
consequences in the short term could be expected to be minimal at the facility and at the Site
boundary.

In the event of the beyond DBE occurring during the period of MCO handling, sampling,
and placement into the storage tubes, the MCOs being handled would be located outside of the
vault. These MCOs would be vulnerable to structural failure caused by the impact of falling
structures, cranes, and other SSCs located in the operating area. In general, at any one time, only
a limited number of MCOs are located at the CSB outside of the storage tubes. As discussed in
Section A3.4.4.1, it is possible to have up to four MCOs outside the storage tubes at the CSB.
These four MCOs would provide the source material for airborne releases and immediate doses to
the public beyond the Site boundary. However, such damaged MCOs would likely have limited
internal pressure with which to cause release of their radioactive particulate inventory, and they
would likely be covered by steel debris from the collapsed operating area shelter, which would
provide limited shielding and airborne pathway disruption. The dose from the facility for this
event is based on the combined airborne particulate released from damaged MCOs. Assuming
four damaged MCOs above the operating deck during the BDBE, the radiological dose to the
onsite receptor would be 1 to 10 rem, which is on the order of four times the consequences of the
mechanical damage to the MCO accident (Section A3.4.2.1).

If the below-grade structures (e.g., operating deck, storage tube assemblies, intake and
exhaust plenums, and MCOs) structurally failed, the SNF would be buried below massive
quantities of concrete and steel debris. There likely would be minimal immediate and short-term
radiological releases and doses from such a catastrophic event. Longer term radiological releases
and doses would depend on the airborne and other release pathways and mechanisms. It is
unlikely that airborne release pathways and airborne dose consequences would be unimpeded and
free flowing over the longer term because the SNF would be contained within steel MCOs and
storage tubes covered by steel and concrete debris.

A3.4.3.3 Beyond Design Basis Flooding of the Underground Vaults. The CSB is at an
elevation of about 700 ft above mean sea level. The MCOs are stored in metal tubes in the CSB
underground vaults approximately 40 ft below grade. The Columbia River is closest to the CSB
at river mile 370 where the normal level is at approximately 375 ft and the estimated flood level,
at an annual probability of $1 \times 10^{-4}$, is 435 ft above mean sea level (WHC-SD-SNF-DB-009).
This is about 265 ft below the CSB. The CSB site is a flood-dry site with respect to river
flooding, so only flooding caused by precipitation runoff need be considered. The probable
maximum precipitation (PMP) for the local storm for CSB is a one-square-mile, 6-hour storm
with 9.2 in. of rainfall for the surrounding watershed runoff (WHC-SD-SNF-DB-009).

The CSB storm drain and flood control components have been designed to accommodate
the PMP event and contain provisions for culverts to allow passage under roadways of runoff
from storms smaller than the PMP. The site civil grading and drainage plan is for the PMP event
and to accommodate for storms larger than the PMP by overflows (larger-than-design flows)
using drain and flood control over the site plan when the system capacity is exceeded. The CSB
criterion for the culverts, drainage ditches, and general storm drainage overflow path is to
maintain safety of the property by ensuring that the PMP event of 9.2 in. of rainfall in 6 hours
does not allow water flow into the CSB. Onsite, runoff water flows to a culvert that passes under
the ramp road immediately northeast of the CSB. This culvert is the only water pathway that
could have any impact on the CSB. In an extreme case, the culvert could be plugged with debris
at the time the PMP storm occurred, backing up rain water and making a pond that would rise up
to the lowest elevation of the road immediately north of the CSB. Other parts of the site drain
away from the CSB. Additional rainfall would increase the depth of flow over the ramped access
road, but the site is designed such that the water level does not build up to a height that would
threaten the CSB, affect its structural integrity, or allow water into the building. The steel cover
over the north intake stack prevents rain from entering the vault through the intake stack.

In a BDBA rainfall or other flooding event, such as a fire main break, water could flow into
the building via doorways, cracks, and other inlets to the building. Water could also enter the
building or the underground vaults via covers to the underground structure adjacent to the intake
stacks on the west side of the building. Water in the vaults should not have structural implications
to the structural integrity of the building nor negative thermal effects on the building structure,
tubes or the MCO shell. Current structural analyses, including CSB-S-0025, HCSA/SNF Vault
Design, and CSB-S-0023, HCSA/SNF Gravity Load Analysis, indicate margins of safety without
seismic loads on the order of 20% to 30%. As an indicative measure, one foot of standing water
imposes a 0.5 lb/in$^2$ load. A catastrophic flood would yield a load on the basemat on the order of
40 lb/in$^2$. This is judged a minimal load on the basemat. For the exterior walls, the soil loads

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imposed on these walls, where the stresses in the walls may change direction, would balance the water loads. For the interior walls, in a catastrophic flood, the loads on the other side of the wall would balance the water loads on one side of the wall by the water in vaults 2 and 3. The N-S end walls would also sustain differential directional loading. In general, the catastrophic flood has minimal structural impact within the existing margins of safety. It is also noted that the heat from the storage tubes and the passive airflow through the vault facilitate vault dryout.

A3.4.4 Common Cause Initiators

The hazard and accident analysis process identified several common cause initiators that have the potential to impact the CSB (HNF-SD-SNF-HIE-001). These initiators have been addressed individually as discrete accidents but are discussed collectively here to address the risks of common cause failure at the CSB.

A3.4.4.1 Seismic Forces. Forces from seismic events could affect the performance of safety SSCs at the CSB during the handling and placement of MCOs or during interim storage of MCOs. Typically, only three MCOs (no more than four) would be handled or processed concurrently in the CSB: one in the receiving pit and two in sampling/weld stations or any two of these locations plus one in the MHM. Equipment handling the MCOs is designed to maintain structural integrity during seismic events and to not fail in such a way as to adversely impact the MCO or to affect other safety function SSCs.

Without controls, a seismic event could lead to handling upsets by cranes, the MHM, and other handling equipment that result primarily in physical impacts to the MCO. In this event, passive features (e.g., cask lifting yoke, limited MHM hoist height, impact absorbers) limit drop heights and provide passive energy absorption and passive barrier protection to the MCO.

A seismic detection system at the CSB disconnects power to the receiving crane and the MHM in the event of a DBE (see Section A2.7.6). On a disconnect of power, the receiving crane load is locked in its current position and the wheel brakes lock, preventing further movement of the load or crane. A disconnect of power to the MHM activates seismic clamps for both sides of the bridge and for both sides of the trolley, stops turret rotation, and locks the shield plug hoist and the MCO hoist in their current positions. The clamps and locking features are designed to prevent further motion of the MHM, minimize damage to the MCO, and prevent potential contamination spread. These actions comply with ASME NOG-1 requirements for power disconnects during a safe-shutdown earthquake.

The sampling equipment and connectors are designed to safety-significant criteria. The sampling/weld station gantry hoist has seismic restraints, and the sampling lines and connectors are designed to withstand the DBE.

The facility structures (vaults, intake structure and exhaust stack, at-grade structures, tube base assemblies, and carbon steel basemat embeds) and storage tube assemblies are passive
safety-class features designed to protect MCOs during the DBE. Thus, releases are not
anticipated during interim storage operations at the CSB, and the controls identified in the DBA
analyses in Section A3.4.2 are sufficient to prevent a release following a DBE.

The CSB has been designed to meet the seismic criteria in WHC-SD-SNF-DB-004 and
DOE-STD-1020-94, Natural Phenomena Hazards Design and Evaluation Criteria for
Department of Energy Facilities, which require that the effects of a 50% greater seismic loading
be no greater than the following (the functions of the structure, including spent fuel cooling,
remain intact).

- Concrete barriers can be cracked, but cracks should be small enough to maintain
  pressure differential with normal HVAC systems. Cracks should not be expected to
  exceed 0.125 in.

- Metal liners and metal pressure-retaining components should remain leak-tight.

- Components should remain anchored in place and be functional.

- Visible local damage occurs, but permanent distortion will not be immediately
  apparent to the naked eye.

A3.4.4.2 Loss of Facility Power. Natural phenomena events (e.g., rain, lightning, freezing
weather) can cause the temporary loss of facility power. In addition, certain external events (e.g.,
vehicle impacts with critical components outside the facility) could also cause a loss of facility
power. There were no loss of power events identified requiring further evaluation (see
HNF-SD-SNF-HIE-001). The facility has a fail-safe design for loss of power to all the SSCs at
the CSB (e.g., MHM [see Chapters A2.0 and A4.0 for details]); and no identified need for active
safety function performance during and after a loss-of-power event. Thus, no releases are
anticipated to occur in the event of a loss of power (see HNF-SD-SNF-HIE-001), and the
features and controls identified in the DBA analyses and discussed in Section A3.4.2 are sufficient
to prevent a release following a loss of power.

A3.4.4.3 High Vault Temperatures. High vault temperatures could lead to a violation of
thermal design criteria for multiple MCOs. High vault temperatures have been analyzed for single
MCOs under unmitigated and mitigated conditions as a DBA in Section A3.4.2.6. The features
identified in the DBA analyses are sufficient to prevent excessive vault temperatures, and
therefore, no releases are expected from such an event.

Furthermore, the effects of volcanic ash that may tend to cling to tube surfaces during a
volcanic eruption have been studied and are documented in Appendix J of CSB-HV-0001,
A Canister Storage Building Vault Thermal Analysis. A conservative condition was assumed:
(1) a 0.25-in. coating of ash with a conductivity coefficient of 0.25 Btu/h-ft-°F adhered to the
vertical tube surfaces and (2) a vault fully loaded with MCOs. This study found that the reduction
in heat transfer led to minor increases in temperatures (on the order of 1 °F to 3 °F in MCO and
concrete temperatures). Thus minimal thermal impacts are expected on peak MCO and concrete
temperatures in the vault. Such volcanic ash conditions should bound the expected impacts from
ash resulting from a range fire because volcanic ash should have larger thermal resistance
properties and densities than ash from a range fire.

A3.4.4.4 Loss of Facility Support Systems. A loss of facility support systems (e.g., HVAC,
power, and inert gas supply) could impact CSB sampling activities if the loss occurred during
sampling of monitored MCOs. Other operational activities are not adversely affected by the loss
of support systems because of the uninterruptible power supply available for monitoring
instrumentation; the fail-safe design of the safety function MHM interlocks at the CSB; and the
lack of need for active safety function performance during and after a loss of HVAC, power, or
inert gas supply. The receiving crane and the MHM do not require power to maintain safe control
of an MCO, do not interface with the HVAC system, and do not utilize inert gas. On a loss of
power, the receiving crane and the MHM stop motion.

The facility HVAC system is not credited to mitigate any radiological releases. However,
the sampling/weld station HVAC system is expected to dilute hydrogen or filter radiological
releases during an MCO sampling accident. In the event of the loss of the inert gas system during
sampling activities, simple operator actions (i.e., closing the sample line valve) put the sampling
configuration into a safe and stable mode. MCO cooling is either designed to be passive or its
absence does not lead to a condition exceeding design criteria for a long time period (months).
Thus, ample time exists for support systems to be reestablished. No releases are anticipated to
occur, and therefore, the features and controls identified in the hazard and DBA analyses are
sufficient to prevent a release caused by the loss of facility support systems.

A3.4.4.5 Facility Fire. The fire hazard analysis (HNF-SD-SNF-FHA-002) identified the CSB
fire-related hazards, preventive and mitigating actions, fire protection features, and damage
potentials from fire. The findings of the fire hazard analysis were incorporated into the design and
engineering of the CSB SSCs and controls. A fire involving a transport vehicle with 100 gal of
fuel had the highest loss. This loss included damage to the trailer vestibule area, damage to
installed equipment, and cleanup of the products of combustion from the operating area. No
specific fire considerations were required for dealing with hydrogen deflagration in the fire hazard
analysis because those considerations are addressed in Sections A3.4.2.3 and A3.4.2.4.

The maximum possible fire loss for the facility was from a support area building fire.
However, this section of the CSB facility has sprinklers to protect it from such damage. The
conclusion of the analysis is that the fire risks present in the CSB facility are acceptable with
regard to protecting employees, property, the public, and the environment from any undue
hazards and that the objectives of DOE Order 5480.7A are met based on the fire protection
features in the CSB (HNF-SD-SNF-FHA-002).

A3.4.4.6 Emergency Evacuation Caused by External Events. If the CSB is evacuated
because of external events at other facilities, a loss of operator control over MCO handling and
monitored MCO sampling could occur. Emergency response procedures will be used to respond to these events; however, the inherent risk is assessed without taking credit for such procedures.

- The cask-MCO can be left unattended on the transporter without risk of radiological consequences because of the passive cooling of the cask-MCO and the structural protection provided to the MCO by the cask.

- Lack of operator control for MHM handling of the MCOs is not a safety concern because of the fail-safe design of the MHM, which, on loss of power or without an operator, will not satisfy permissives and will shut down with no ability for motion.

- During sampling operations, the sampling operator has the simple action of closing the MCO valve operator or the sampling valve before evacuation.

Consequently, evacuation due to external events is not anticipated to lead to events with radiological releases and is not a safety concern. A facility evacuation due to internal events is adequately protected against with the existing features and controls identified within the DBAs. Therefore, the facility SSCs and controls are adequate to passively maintain safe conditions for all MCOs during and after an evacuation.

A3.5 REFERENCES


ACI-349, 1991, Code Requirements for Nuclear Safety Related Concrete Structures, American Concrete Institute, Farmington Hills, Michigan.


Annex A — Canister Storage Building


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APPENDIX A3A

ABNORMAL EVENT BINS FOR THE CANISTER STORAGE BUILDING
**Table A3A-1. Abnormal Event Bins for the Canister Storage Building.** (5 sheets)

<table>
<thead>
<tr>
<th>Event</th>
<th>Means of detection</th>
<th>Consequences</th>
<th>Corrective actions*</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Bin 1: MCO in a &quot;suspect&quot; condition</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Entry condition (MCO received)</td>
<td>- Operator observation</td>
<td>- Facility delay for investigation</td>
<td>- Verify MCO confinement integrity</td>
</tr>
<tr>
<td></td>
<td>- Pressure</td>
<td>- Facility delay for special handling</td>
<td>- Isolate the MCO from the facility atmosphere (install tent or &quot;greenhouse,&quot; or place MCO in an overpack tube)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Increased occupational radiation exposure</td>
<td>- Establish monitoring routine</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Delay at other facilities</td>
<td>- Cask service system, pressure safety devices</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- Sample and reinert MCO as necessary</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- Maintain other MCOs not in storage in safe and stable conditions, or place in storage</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>- Notify other SNF Project facilities of potential process impact</td>
</tr>
<tr>
<td>MCO impacts</td>
<td>Operator observation</td>
<td>- Facility delay for investigation</td>
<td>- Verify MCO confinement integrity</td>
</tr>
<tr>
<td></td>
<td></td>
<td>- Facility delay for special handling</td>
<td>- Isolate the MCO from the facility atmosphere (install tent or &quot;greenhouse,&quot; place MCO in an overpack tube)</td>
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<td></td>
<td></td>
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</tbody>
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<table>
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<tr>
<th>Event</th>
<th>Means of detection</th>
<th>Consequences</th>
<th>Corrective actions*</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>MCO mishandling or misloading</strong></td>
<td>Operator observation</td>
<td>• Facility delay for investigation</td>
<td>• Verify MCO confinement integrity</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Facility delay for special handling</td>
<td>• Isolate the MCO from the facility atmosphere (install tent or &quot;greenhouse,&quot; place MCO in an overpack tube)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Increased occupational radiation exposure</td>
<td>• Establish monitoring routine</td>
</tr>
<tr>
<td></td>
<td></td>
<td>• Delay at other facilities</td>
<td>• Sample and reinert MCO as necessary</td>
</tr>
<tr>
<td></td>
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<td></td>
<td>• Maintain other MCOs not in storage in safe and stable conditions, or place in storage</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Notify other SNF Project facilities of potential process impact</td>
</tr>
</tbody>
</table>

**Bin 2: Loss of ability to normally handle or store cask-MCO**

2A: Electrical/mechanical failure

- Loss of power
- MHM mechanical breakdown
- Handling systems, mechanical breakdown
- Cranes, breakdown or collisions

- Alarm
- Operator observation

- Facility schedule delay
- Increased occupational radiation exposure
- Project schedule delay

- Verify MCO confinement integrity for MCOs not in storage
- If the event causes a loss of power, establish loss-of-power breaker alignment to prevent spurious equipment operation when power is restored, and reestablish normal power supplies following return of power
- Isolate damaged equipment
- Transition MCOs not in storage to safe and stable positions, including manual positioning during a loss-of-power event
- Maintain other MCOs not in storage in safe and stable conditions, or place in storage
- Notify other SNF Project facilities of potential process impact
<table>
<thead>
<tr>
<th>Event</th>
<th>Means of detection</th>
<th>Consequences</th>
<th>Corrective actions*</th>
</tr>
</thead>
<tbody>
<tr>
<td>Vehicle collisions</td>
<td>Alarm</td>
<td>Facility schedule delay</td>
<td>• Verify MCO confinement integrity for MCOs not in storage</td>
</tr>
<tr>
<td>Natural phenomena structural impacts on CSB</td>
<td>Operator observation</td>
<td>Increased occupational radiation exposure</td>
<td>• If the event causes a loss of power, establish loss-of-power breaker alignment to prevent spurious equipment operation when power is restored, and reestablish normal power supplies following return of power</td>
</tr>
<tr>
<td>Natural phenomena impacts on CSB operations</td>
<td>Notification</td>
<td>Project schedule delay</td>
<td>• Isolate damaged equipment</td>
</tr>
<tr>
<td>Leak of water into building</td>
<td></td>
<td></td>
<td>• Erect temporary barriers, if external walls are damaged</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Control water source, erect dams to divert water away from vault and operating bay, and initiate action to remove residual water</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Transition MCOs not in storage to safe and stable positions, including manual positioning during a loss-of-power event or if crane damage occurs</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Maintain other MCOs not in storage in safe and stable conditions, or place in storage</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Notify other SNF Project facilities of potential process impact</td>
</tr>
</tbody>
</table>
Table A3A-1. Abnormal Event Bins for the Canister Storage Building. (5 sheets)

<table>
<thead>
<tr>
<th>Event</th>
<th>Means of detection</th>
<th>Consequences</th>
<th>Corrective actions*</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fire</td>
<td>• Alarm</td>
<td>• Facility schedule delay</td>
<td>• Verify MCO confinement integrity for MCOs not in storage near the fire</td>
</tr>
<tr>
<td></td>
<td>• Operator observation</td>
<td>• Increased occupational radiation exposure</td>
<td>• If fire causes loss of power, establish loss-of-power breaker alignment to prevent spurious equipment operation when power is restored, and reestablish normal power supplies following return of power</td>
</tr>
<tr>
<td></td>
<td>• Notification</td>
<td>• Project schedule delay</td>
<td>• Isolate equipment involved in the fire</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• If water was used to extinguish the fire, determine effect on facility safety limits</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• If water was used, immediately establish dams to divert water away from the vault and operating bay, and initiate action to remove the residual water</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Transition MCOs not in storage to safe and stable positions, including manual positioning during a loss of power or if crane damage occurs</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Maintain other MCOs not in storage in safe and stable conditions, or place in storage</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>• Notify other SNF Project facilities of potential process impact</td>
</tr>
</tbody>
</table>
### Table A3A-1. Abnormal Event Bins for the Canister Storage Building. (5 sheets)

<table>
<thead>
<tr>
<th>Event</th>
<th>Means of detection</th>
<th>Consequences</th>
<th>Corrective actions*</th>
</tr>
</thead>
<tbody>
<tr>
<td>Operator errors</td>
<td>• Alarm</td>
<td>• Facility schedule delay</td>
<td>• Verify MCO confinement integrity for MCOs not in storage</td>
</tr>
<tr>
<td></td>
<td>• Operator observation</td>
<td>• Increased occupational radiation exposure</td>
<td>• If the event causes a loss of power, establish loss-of-power breaker alignment to prevent spurious equipment operation when power is restored, and reestablish normal power supplies following return of power</td>
</tr>
<tr>
<td></td>
<td>• Notification</td>
<td>• Project schedule delay</td>
<td>• Isolate damaged equipment</td>
</tr>
</tbody>
</table>

*Actions to be evaluated during recovery

CSB = Canister Storage Building.
HVAC = heating, ventilation, and air conditioning.
MCO = multi-canister overpack.
MHM = multi-canister overpack handling machine.
SNF = spent nuclear fuel.

Bin 2D: Errors

- Transition MCOs not in storage to safe and stable positions, including manual positioning if necessary
- Maintain other MCOs not in storage in safe and stable conditions, or place in storage
- Notify other SNF Project facilities of potential process impact
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CHAPTER A4.0

SAFETY STRUCTURES, SYSTEMS, AND COMPONENTS
NOTE: If this information is being provided for the purposes of evaluating the multi-canister overpack handling machine (MHM) and its components under the realm of the Spent Nuclear Fuel Project, limited use restrictions apply to the documents prepared by GEC Alsthom Engineering Systems Ltd. that are referenced in this appendix. “Limited rights data” requirements and use are explained below.

Further disclosure or use of GEC Alsthom Engineering Systems’ technical data may be made for the following purposes.

1. The “limited rights data” may be disclosed for evaluation purposes under the restriction that the “limited rights data” be retained in confidence and not be further disclosed.

2. The “limited rights data” may be disclosed to other Contractors participating in the Government’s program of which the MHM contract is a part for information or use in connection with the work performed under a government contract and under the restriction that the “limited rights data” be retained in confidence and not be further disclosed.

3. The “limited rights data” may be used by the Government or others on its behalf for emergency repair or overhaul work under the restriction that the “limited rights data” be retained in confidence and not be further disclosed.
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<th>Definition</th>
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<tr>
<td>AC</td>
<td>Administrative Control</td>
</tr>
<tr>
<td>ALARA</td>
<td>as low as reasonably achievable</td>
</tr>
<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
</tr>
<tr>
<td>DBA</td>
<td>design basis accident</td>
</tr>
<tr>
<td>DBE</td>
<td>design basis earthquake</td>
</tr>
<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
</tr>
<tr>
<td>FFTF</td>
<td>Fast Flux Test Facility</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>HEPA</td>
<td>high-efficiency particulate air (filter)</td>
</tr>
<tr>
<td>HVAC</td>
<td>heating, ventilation, and air conditioning</td>
</tr>
<tr>
<td>ITS</td>
<td>important to safety</td>
</tr>
<tr>
<td>LCO</td>
<td>Limiting Condition for Operation</td>
</tr>
<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
</tr>
<tr>
<td>MHM</td>
<td>multi-canister overpack handling machine</td>
</tr>
<tr>
<td>NPH</td>
<td>natural phenomena hazard</td>
</tr>
<tr>
<td>NRC</td>
<td>U.S. Nuclear Regulatory Commission</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
</tr>
<tr>
<td>SRSS</td>
<td>square root of the sum of the squares</td>
</tr>
<tr>
<td>SSC</td>
<td>structure, system, and component</td>
</tr>
<tr>
<td>SSI</td>
<td>soil-structure interaction</td>
</tr>
<tr>
<td>TSR</td>
<td>technical safety requirement</td>
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A4.0 SAFETY STRUCTURES, SYSTEMS, AND COMPONENTS

A4.1 INTRODUCTION

This chapter provides details on those facility structures, systems, and components (SSCs) that are necessary for the facility to satisfy evaluation guidelines, provide defense in depth, contribute to worker safety, or meet important-to-safety criteria. Descriptions of the attributes (i.e., functional requirements and performance criteria) required to support the safety functions identified in the hazard and accident analyses and to support subsequent derivation of technical safety requirements (TSRs) are provided. This chapter meets the requirements of DOE Order 5480.23, Nuclear Safety Analysis Reports, follows the format guidance in DOE-STD-3009-94, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports, and includes the following information.

- A list of the safety-class SSCs and subsections containing details of safety functions, a system description, functional requirements, a system evaluation, and a list of assumptions requiring TSRs to ensure performance of the safety function.

- A list of the safety-significant SSCs and subsections containing details of safety functions, a system description, functional requirements, a system evaluation, and a list of assumptions requiring TSRs to ensure performance of the safety function.

The Canister Storage Building (CSB) [Building 212H] has been designed using an SSC safety classification based on Letter 97-SFD-172, Risk Evaluation Guidelines (REGs) to Ensure Inherently Safer Designs (Sellers 1997), and on HNF-PRO-704, Hazard and Accident Analysis Process. HNF-PRO-704 implements the safety classification methodology presented in DOE Order 6430.1A, General Design Criteria, for safety-class SSCs and in Letter 97-SFD-172 (Sellers 1997) for safety-significant SSCs. A graded approach has been used based on the unmitigated consequences of potential accidents. SSCs are designated as safety class or safety significant based on their safety function for the prevention or mitigation of potential accidents or to ensure that the assumptions made in safety analyses remain valid.

In general, safety-class SSCs are those items required for protection of the offsite public. Safety-class SSCs include those items designated as safety class in accordance with DOE Order 6430.1A. Safety-class SSCs also encompass items that are designated as “important to safety” (ITS) in accordance with Title 10, Code of Federal Regulations, Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste,” Section 72.3, “Definitions” (10 CFR 72), and have been classified as Category A as defined in Item 29 of HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements.

Safety-significant items are those SSCs required for the protection of onsite personnel not directly involved in facility operations. Safety-significant SSCs also encompass those items that
have been designated ITS Category B in accordance with Item 29 of HNF-SD-SNF-DB-003. SSCs that would prevent a fatality or protect multiple facility workers from serious injury (not standard industrial hazards), or those SSCs that would prevent or mitigate toxic chemical exposures, also are designated safety significant. Those SSCs whose failure could impact or impair the function of safety SSCs are designated as the same level of importance as the potentially affected SSCs, except for the operating shelter, receiving crane, and the multi-canister overpack handling machine (MHM). Although the operating shelter, receiving crane, and the MHM are designated as safety-significant, they have been designed to safety-class design basis earthquake (DBE) loads and analyzed to this higher loading to ensure adequate margins of safety to prevent collapse of these structures onto the safety-class operating deck during a DBE. Both safety-class and safety-significant SSCs are subject to higher control levels and reviews than general service SSCs during design, fabrication, procurement, and installation.

All SSCs that are not classified as safety class or safety significant are general service SSCs. Those SSCs that have been designated as ITS Category C also are included in the general service SSCs. These SSCs protect workers from standard industrial hazards or are controlled by site safety programs. Items that do not perform a safety function other than industrial safety or that have a minor impact on safety are designated general service SSCs following the guidance in HNF-PRO-704.

### A4.2 REQUIREMENTS

This section provides a summary list of the U.S. Department of Energy (DOE) orders and standards, U.S. Nuclear Regulatory Commission (NRC) rules and guidance, industry standards and design codes, and Hanford Site requirements used in establishing the facility safety basis. The intent is to provide only the requirements that are specific to this chapter and pertinent to the safety basis.

#### A4.2.1 U.S. Department of Energy Orders and Standards

The following DOE orders and standards are applicable to the facility safety basis.

- **DOE Order 6430.1A, General Design Criteria.** Compliance to this order is documented in FDH-9951266, *Canister Storage Building 6430.1A and Nuclear Regulatory Commission Equivalency Compliance Matrix* (Williams 1999), which provides a requirement-by-requirement evaluation of the compliance of the CSB design to DOE Order 6430.1A. Attachment 3 of FDH-9951266 contains a list of areas where further action is needed to demonstrate compliance. These areas are primarily in startup and test, and acceptance testing.
• DOE Order 5480.24, Nuclear Criticality Safety. This order establishes nuclear
criticality standards and criteria for nuclear facilities. Additional discussion of the
CSB nuclear criticality safety criteria is provided in Chapter A6.0.

• DOE Order 5480.28, Natural Phenomena Hazards Mitigation. This order
establishes mitigation requirements for natural phenomena hazards (NPHs) and
targets probabilistic performance goals based on the facility performance category.
Additional discussion of the NPH performance criteria is provided in Section A1.5.
DOE Order 5480.28 is implemented via WHC-SD-SNF-DB-009, Canister Storage
Building Natural Phenomena Hazards.

• DOE-STD-1020-94, Natural Phenomena Hazards Design and Evaluation Criteria
for Department of Energy Facilities. This standard provides the methodology
applicable to design of nuclear facilities for NPH events.

A4.2.2 U.S. Nuclear Regulatory Commission Rules and Guidance

The following NRC rules and guides are applicable to the facility safety basis.

• Title 10, Code of Federal Regulations, Part 72, “Licensing Requirements for the
Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste”
(10 CFR 72). The design bases for items identified as ITS must reflect appropriate
combinations of the effects of normal and accident conditions and the effects of
natural phenomena. This is implemented by the application of the design criteria for
safety-class items, as provided in DOE Order 6430.1A, to those items identified as
ITS. Applying NRC equivalency requires that SSCs ITS be identified in accordance
with the definition in 10 CFR 72.3. Once SSCs ITS have been identified, a graded
approach is applied using the definitions in HNF-SD-SNF-DB-003. ITS is
subdivided into the categories defined in HNF-SD-SNF-DB-003. SSCs found to
belong in categories A or B must also meet 10 CFR 72, Subparts F and G, using a
graded approach.

  – Category A — Critical to Safe Operation

  SSCs in this category include those whose failure or malfunction could
directly result in a condition adverse to public health and safety. ITS SSCs in
this category have been classified as safety class using the definition in DOE
Order 6430.1A, with the additional requirements therein.

  – Category B — Major Impact on Safety

  SSCs in this category include those whose failure or malfunction could result
in a condition adversely affecting collocated worker health and safety.
Note that from the definition of Category C, Category B is understood to include events that could significantly damage the multi-canister overpack (MCO) without severe impact to public health and safety. SSCs in this category are classified as safety significant.

- **Category C — Minor Impact on Safety**

SSCs whose failure or malfunction would not significantly reduce containment and would not be likely to create a situation adversely affecting public or collocated workers’ health and safety. SSCs in this category are classified as general service (safety class 3 [nonsafety] in DOE Order 6430.1A).

The SSCs identified as ITS in accordance with 10 CFR 72.3 are listed in Tables A4-1 and A4-9. These lists also contain the ITS classification for each SSC as determined by Chapter A3.0. The ITS classification is directly incorporated into the tables of preventive and mitigative safety function features identified at the end of each design basis accident (DBA) scenario in Section A3.4.2. Three hoists, the tent gantry hoist and the two sampling/weld station hoists, are the only general service equipment items that are classified as ITS Category C because the hoists can potentially drop equipment that could cause minor damage to the top of an MCO.

Additional discussion of the NRC criteria for mitigation of natural phenomena is provided in Section A1.2.

- **NRC Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants.** This regulatory guide is invoked by HNF-SD-SNF-DB-003. For the CSB, the guide is used to assist in assigning the appropriate code class to systems and components designed to the Boiler and Pressure Vessel Code (ASME 1995), Section III. This guidance was used in selecting the Boiler and Pressure Vessel Code (ASME 1995), Section III, Subsection NC, as the appropriate code for the overpack storage tubes and the overpack storage tube plugs.

- **NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.** This plan provides guidance that is used in WHC-SD-SNF-DB-009 to establish probabilistic risk assessment techniques for tornado-generated missile protection of specific targets.
A4.2.3 Industry Consensus Standards and Other Documentation

The following industry standards are applicable to the safety basis for the facility.

- ANSI N14.6-1993, Radioactive Materials — Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More. This standard provides design criteria for special lifting devices for critical loads and is used in the design of the MHM MCO grapple, the standard and overpack tube plug pintle, and the receiving crane lifting yoke. Compliance with the added safety margins required by this standard provides a high degree of assurance against inadvertently dropped loads.

- ANSI/ANS-57.9-1992, Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type). This standard includes requirements for the design of major buildings and structures, including physical security and safety features for independent spent fuel storage installations. An application evaluation against the requirements of ANSI/ANS-57.9-1992 was included in Addendum 3 of Attachment 1 to FDH-9761261 R4, Safety Classification of Cranes and Handling Equipment (Williams 1998b). The information included a listing of those items cited in ANSI/ANS-57.9-1992 as relevant to nuclear safety that are not in the CSB design, justification for not including them, and a listing of the items cited in ANSI/ANS-57.9-1992 that were included as criteria independent of ANSI/ANS-57.9-1992. An earlier review of ANSI/ANS-57.9-1992 identified an additional requirement from Section 6.17.1.1(1) to increase the dead load of structures by 5%. The 5% dead load contingency has been included in the design of the CSB. In Letter FDH-9755210 R1, Response to Issues Associated with Fabrication Release for the Spent Nuclear Fuel Project Multi-Canister Overpack Handling Machine (Mahaffey 1997), the MHM design was found to be consistent with the requirements of ANSI/ANS-57.9-1992. An updated ANSI/ANS-57.9-1992 compliance matrix, Attachment 5 of SNF-5790, Design Compliance Matrices to ANSI and OSHA, was prepared and provides justification that the CSB complies with the criteria in ANSI/ANS-57.9-1992. Design compliance statements for the receiving crane and the MHM were provided for all issues and additional requirements identified in the earlier evaluation.

- ANSI/ACI 349-85, Code Requirements for Nuclear Safety Related Concrete Structures. The requirements of ANSI/ACI 349-85 were applied to the design of safety-class and safety-significant reinforced concrete structures following the guidance in HNF-PRO-097, Engineering Design and Evaluation, and FRP-112, SNF Canister Storage Building (Bedell 1996a; this is the architect-engineer's Design Basis Document).

- ANSI/AISC N690-94, Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities. The requirements of
ANSI/AISC N690-94 were applied to the design of safety-class and safety-significant steel structures following the guidance provided in HNF-PRO-097 and the architect-engineer's Design Basis Document (Bedell 1996a).

- ASME N509-1989, Nuclear Power Plant Air-Cleaning Units and Components New Construction and Modification Only, and ASME N510-1989, Testing of Nuclear Air-Cleaning Systems. These two standards were used in the design of the CSB high-efficiency particulate air (HEPA) filter enclosures.

- ANSI/ANS-57.1-1992, Design Requirements for Light Water Reactor Fuel Handling System. According to HNF-SD-SNF-DB-003, the requirements in ANSI/ANS-57.1-1992 for cranes and lifting devices used in fuel handling operations are to be incorporated into the design of the CSB. A review of the MHM and receiving crane design packages concluded that some of the NRC requirements identified already were included as features in the vendor designs either because they were included in the procurement documents or were required by ASME NOG-1-1995, Rules for the Construction of Overhead and Gantry Cranes.

- ANSI/ANS-57.2-1992, Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants. According to HNF-SD-SNF-DB-003, the requirements in ANSI/ANS-57.2-1992 for cranes and lifting devices used in spent fuel handling operations are required to be incorporated into the design of the CSB. FDH-9855462 (Williams 1998a) evaluates compliance with and deviations from the requirements of ANSI/ANS-57.1-1992, Section 5.0, “System Functional Description,” and Section 6.0, “Design Requirements.” Certain of these requirements apply to the design of the CSB. A later evaluation (Williams 1998b) identified applicable requirements from this standard for the MHM and receiving crane. Part of the requirements was already in the designs, and the remainder was added using the Design Change Notice process. The evaluation concluded that additional modifications of the project cranes are not necessary and that the NRC equivalency requirement, Item 13, is met.
• ASCE-4-86, *Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety-Related Nuclear Structures*. ASCE-4-86 provides acceptable methods for seismic analyses, which were used for analysis of safety-class and safety-significant structures. Application of this standard conforms with the guidance provided in DOE-STD-1020-94, HNF-PRO-097, and the architect-engineer's Design Basis Document (Bedell 1996a).

• ASCE-7-93, *Minimum Design Loads for Building and Other Structures*. ASCE-7-93 provides for snow and wind loads to be applied to the design of structures. This standard is used in the Design Basis Document (Bedell 1996a) for load combinations with wind and snow loads following the guidance in HNF-PRO-097 and Table A1-1.

• ASME B30.20-1993, *Below-the-Hook Lifting Devices*. ASME B30.20-1993 provides requirements for the design and fabrication of lifting devices that attach to a crane lifting hook. This specification is used to establish the requirements for the receiving crane transport cask lifting yoke and the MHM tube and shield plug grapples.

• ASME *Boiler and Pressure Vessel Code* (ASME 1995). Safety-class vessels are designed in accordance with the requirements of Section III, Class 2, of the *Boiler and Pressure Vessel Code*. Pressure vessels designated safety significant and general service are designed in accordance with the requirements of Section VIII, Division 1. Application of the *Boiler and Pressure Vessel Code* is required by HNF-SD-SNF-DB-003, in conjunction with NRC Regulatory Guide 1.26.

• ASME NOG-1-1995, *Rules for the Construction of Overhead and Gantry Cranes*. ASME NOG-1-1995 provides requirements for the design, fabrication, and erection of overhead cranes used in nuclear facilities. The ASME NOG-1-1995 design requirements for Type I cranes are imposed on the receiving crane and the MHM crane gantry and hoists by their respective performance specifications. This specification also is used for seismic analysis and design of the MHM and receiving crane.

• NRC Regulatory Guide 1.84, *Design and Fabrication Code Case Acceptability — ASME Section III, Division 1*, and NRC Regulatory Guide 1.85, *Materials Code Case Acceptability — ASME Section III, Division 1*, were reviewed to determine NRC positions on the *Boiler and Pressure Vessel Code* (ASME 1995), Section III code cases for safety class I applications. After reading the titles and brief descriptions, four code cases were investigated. These cases involved alternate rules for examination of welds (Regulatory Guide 1.85), a certified design report summary for component standard supports (Regulatory Guide 1.84), tack welds for major components and piping supports (Regulatory Guide 1.84) and major component
material requirements (Regulatory Guide 1.85). No additional requirements applicable to the CSB were identified from this review of Regulatory Guides 1.84 and 1.85. These code cases were not used in the design of the Boiler and Pressure Vessel Code (ASME 1995), Section III components because the design requirements for the components are adequate and provide similar safety functions. Therefore, the NRC equivalency requirements, Item 11, for the Boiler and Pressure Vessel Code, Section III (ASME 1995), are met.


A4.2.4 Hanford Site Requirements

The following Hanford Site standards are applicable to the safety basis for the facility. Implementation of these standards ensures consistent compliance with DOE orders and implementing standards.

- WHC-CM-4-46, Safety Analysis Manual, Revision 1, as modified by Letter 96-SFD-318, Contract No. DE-AC06-96RL13200: Proposed Spent Nuclear Fuel Project Division (SFD) Risk Acceptance Guidelines (RAGs) (Sellers 1996). Chapter 9.0 of WHC-CM-4-46 identifies the methodology for determining the safety classification of SSCs and the applicable design criteria. The safety-class requirements in Chapter 9.0 of WHC-CM-4-46 implement the criteria established in DOE Order 6430.1A for safety-class items and the requirements for 10 CFR 72 ITS SSCs. Design criteria for safety-significant SSCs also are provided in Chapter 9.0 of WHC-CM-4-46.

The previous versions of the CSB safety analysis report were originally written under a waiver to Chapter 9.0, Revision 2, of WHC-CM-4-46; and the last revision of the phased safety analysis report began incorporating the safety-class/safety-significant terminology of Revision 2. Both sets of terminology were used in the text of this chapter in the last phased safety analysis report revision. This final safety analysis report (FSAR) is written using the guidance of HNF-PRO-704.

- HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements. This document provides guidance on demonstration of NRC equivalency. The major NRC equivalency items related to the CSB design and construction are the seismic criteria contained in WHC-SD-SNF-DB-004, Spent Nuclear Fuel Project Seismic Design Criteria, NRC Equivalency Evaluation Report. Other NPHs required for consideration by HNF-SD-SNF-DB-003 are
described in WHC-SD-SNF-DB-009 and included in Table A1-1. Other impacts to
the design caused by NRC requirements are the use of ANSI/ANS-57.9-1992 for the
application of 5% dead load contingency and the use of ANSI/ANS-57.1-1992 for
lifting requirements for the receiving crane and MHM.

- WHC-SD-SNF-DB-004, 1996, Spent Nuclear Fuel Project Seismic Design Criteria,
  NRC Equivalency Evaluation Report. WHC-SD-SNF-DB-004 provides the basis
  for the Spent Nuclear Fuel (SNF) Project decision and justification that the criteria
  of DOE Order 5480.28 and its implementing standards, and the resulting
  performance category 3, provide adequate seismic design safety. Safety-class SSCs
  are designed to the seismic spectra curves contained in this document and included
  in Figure A4-1.

- WHC-SD-SNF-DB-009, 1996, Canister Storage Building Natural Phenomena
  Hazards. WHC-SD-SNF-DB-009 supplements the Performance Category-3 loads
  in DOE-STD-1020-94 and HNF-PRO-097 for use in the design and construction of
  the CSB. Compliance to both NRC and DOE requirements is accomplished by
designing the CSB to the higher loads recommended by this document. Applicable
  loads of WHC-SD-SNF-DB-009 for safety-class SSCs are contained in Table A1-1.

- HNF-S-0425, Performance Specification for the Spent Nuclear Fuel Canister
  Storage Building. HNF-S-0425 requires that the CSB implement the spectra given
  in DOE-RL-HPS-SDC-4.1, Standard Architectural-Civil Design Criteria: Design
  Loads for Facilities, in designing the facility for a DBE. Earthquake time histories
  for the DBE may be used to satisfy these loading requirements for design and
  analysis of safety-class nonreactor nuclear systems and components as defined in
  DOE-RL-HPS-SDC-4.1.

These loading requirements are supplemented by WHC-SD-SNF-DB-004 as
required by WHC-SD-SNF-DB-009 to comply with NRC equivalency criteria.
Safety-significant items shall be designed to withstand wind loading and
wind-generated missiles according to HNF-PRO-097. Features necessary to protect
the offsite and onsite personnel shall be designed to withstand design basis wind and
missiles, with the criteria for high-hazard usage applicable to safety-class items and
moderate hazard usage applicable to safety-significant items.

- HNF-PRO-097, Engineering Design and Evaluation. This procedure provides
criteria for the design of the CSB. This document is invoked by HNF-S-0425 and
HNF-PRO-704. Section 5.0 of this procedure provides structural design criteria
loads and load combinations for SSCs in Performance Category-4, 3, 2, 1 facilities.
These include dead, live, snow, and soil loads; normal operating loads; and natural
phenomena loading of extreme wind, earthquake, ashfall, and flood. The facility
design has been evaluated for applicable natural phenomena loads listed on
Table A1-1 of Section A1.5, “Natural Phenomena Threats,” for the SSCs impacted.
These loads are higher than HNF-PRO-097 loads, particularly those associated with tornado wind and probable maximum precipitation effects. For example, the tornado wind speed from Table A1-1 is given as 200 mi/h total, versus no tornado wind speed in HNF-PRO-097. This higher wind speed on the operating area shelter transmits higher loads to the below-grade portion of the CSB. The design of the CSB takes into account other NPH loads from Table A1-1. For example, a ground snow load of 20 lb/ft² in Table A1-1 is in excess of the HNF-PRO-097 loading of 15 lb/ft².

A4.2.5 Fluor Daniel, Incorporated, Requirements

- FRP-112, SNF Canister Storage Building (Bedell 1996a). FRP-112 provides information on the design intent, the design methods, and the functions to be performed by SSCs within the CSB. This Design Basis Document (Bedell 1996a) supplements the SNF CSB Performance Specification and provides more detail but does not subtract from the DOE or additional NRC equivalency requirements (HNF-SD-SNF-DB-003). Design of safety-class and safety-significant SSCs is performed to the criteria contained in this document.

A4.3 SAFETY-CLASS STRUCTURES, SYSTEMS, AND COMPONENTS

The safety-class SSCs are summarized in Table A4-1. The table also presents the ITS classification of the SSCs. These safety-class SSCs are designed to the criteria of DOE Order 6430.1A and HNF-PRO-097, applying NPH loads from Section A1.5 for the postulated accident criteria. The subsections of Section A4.3 provide details expanding on the summary descriptions in Table A4-1. The safety significant-functions of safety-class equipment are also shown in Table A4-1.

Nuclear safety equivalency to comparable NRC-licensed facilities is implemented by compliance with existing and applicable DOE requirements coupled with the additional requirement items documented in HNF-SD-SNF-DB-003 that are applicable to the CSB. Compliance with the additional HNF-SD-SNF-DB-003 requirements is documented in a compliance matrix (HNF-4776) and summarized in Table A4-2.

A4.3.1 Canister Storage Building Subsurface Structure

A4.3.1.1 Safety Function. The safety function of the CSB subsurface structure is to prevent damage to MCOs caused by structural failure and provide for natural convective cooling of MCOs that contain SNF until their final disposition has been determined. This forms the design life of the CSB including the subsurface structures as 40 years. The CSB subsurface structure consists of the vault’s concrete basemat, exterior perimeter and interior partition walls, and air
<table>
<thead>
<tr>
<th>Safety class SSC (Chapter A4.X section)</th>
<th>Accident section (Chapter A3.X section)</th>
<th>Safety function</th>
<th>Functional requirements</th>
<th>Performance criteria (TSR applicability, Chapter A5.X section)</th>
<th>ITS category</th>
</tr>
</thead>
<tbody>
<tr>
<td>CSB subsurface structure (vaults, air intake, and exhaust plena) (A4.3.1)</td>
<td>A3.4.2.1</td>
<td>Prevent damage to MCO caused by structural failure</td>
<td>Withstand all DBA structural loading conditions</td>
<td>Configuration control (A5.3.2.3, A5.6.3)</td>
<td>A</td>
</tr>
<tr>
<td></td>
<td>A3.4.2.1</td>
<td>Prevent mechanical damage of MCO</td>
<td>Maintain geometry assumptions used in analyses</td>
<td></td>
<td></td>
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<tr>
<td></td>
<td>A3.4.2.6</td>
<td>Provide for natural convective cooling</td>
<td>Maintain geometry to allow for cooling</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>A3.4.4</td>
<td>Prevent damage to MCO caused by structural failure</td>
<td>Withstand all NPH structural loading conditions</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Carbon steel basemat embeds (A4.3.2)</td>
<td>A3.4.2.1</td>
<td>Maintain horizontal stability of standard and overpack storage tubes</td>
<td>Withstand all DBA structural loading conditions</td>
<td>Configuration control (A5.3.2.4, A5.6.4)</td>
<td>A</td>
</tr>
<tr>
<td></td>
<td>A3.4.2.6</td>
<td>Provide for natural convective cooling</td>
<td>Maintain geometry to allow for cooling</td>
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<tr>
<td></td>
<td>A3.4.4</td>
<td>Prevent damage to MCO caused by structural failure</td>
<td>Withstand all NPH structural loading conditions</td>
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</tr>
<tr>
<td>CSB at-grade structures (operating area deck [including the sample/weld area, and the load-in/load-out area] bases for intake and exhaust structures) (A4.3.3)</td>
<td>A3.4.2.1</td>
<td>Prevent damage to MCO caused by structural failure</td>
<td>Withstand all DBA structural loading conditions</td>
<td>Configuration control (A5.3.2.5, A5.6.5)</td>
<td>A (except support area building foundation, which is NA)</td>
</tr>
<tr>
<td></td>
<td>A3.4.2.6</td>
<td>Prevent mechanical damage of MCO</td>
<td>Maintain geometry assumptions used in analyses</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>A3.4.4</td>
<td>Provide for natural convective cooling</td>
<td>Maintain geometry to allow for cooling</td>
<td></td>
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</tr>
<tr>
<td></td>
<td>A3.4.4</td>
<td>Prevent damage to MCO caused by structural failure</td>
<td>Withstand all NPH structural loading conditions</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Safety class SSC (Chapter A4.X section)</td>
<td>Accident section (Chapter A3.X section)</td>
<td>Safety function</td>
<td>Functional requirements</td>
<td>Performance criteria (TSR applicability, Chapter A5.X section)</td>
<td>ITS category</td>
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<tr>
<td>Standard storage tubes and tube base assemblies (A4.3.4)</td>
<td>A3.4.2.1</td>
<td>Prevent damage to MCO caused by structural failure</td>
<td>Withstand all DBA structural loading conditions</td>
<td>Configuration control (A5.6.6)</td>
<td>A</td>
</tr>
<tr>
<td></td>
<td>A3.4.2.6</td>
<td>Prevent mechanical damage of MCO</td>
<td>Maintain geometry assumptions used in analyses</td>
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<tr>
<td></td>
<td>A3.4.4</td>
<td>Provide for natural convective cooling</td>
<td>Maintain geometry to allow for cooling</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Overpack storage tubes and tube base assemblies (A4.3.5)</td>
<td>A3.4.2.1</td>
<td>Prevent mechanical damage of MCO</td>
<td>Absorb impact loading from dropped MCOs</td>
<td>Configuration control (A5.6.7)</td>
<td>A</td>
</tr>
<tr>
<td></td>
<td>A3.4.2.6</td>
<td>Maintain geometric configuration for storage of MCOs</td>
<td>Maintain geometry assumptions used in analyses</td>
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<tr>
<td></td>
<td>A3.4.4</td>
<td>Provide for natural convective cooling</td>
<td>Maintain geometry to allow for cooling</td>
<td></td>
<td></td>
</tr>
<tr>
<td>CSB intake structure and exhaust stack (A4.3.6)</td>
<td>A3.4.2.6</td>
<td>Prevent damage to MCO caused by structural failure</td>
<td>Withstand all NPH structural loading conditions</td>
<td>Configuration control (A5.3.2.5, A5.6.5)</td>
<td>A</td>
</tr>
<tr>
<td></td>
<td>A3.4.4</td>
<td>Provide passive cooling for MCOs</td>
<td>Maintain MCO external temperature below 270 °F</td>
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<td></td>
<td></td>
<td></td>
<td>Maintain vault concrete temperature below 150 °F</td>
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</table>

March 2000
<table>
<thead>
<tr>
<th>Safety class SSC section</th>
<th>Accident section section</th>
<th>Safety function</th>
<th>Functional requirements</th>
<th>Performance criteria (TSR applicability, Chapter A5.X section)</th>
<th>ITS category</th>
</tr>
</thead>
<tbody>
<tr>
<td>MCO (A4.3.7)</td>
<td>A3.4.2.1</td>
<td>Maintain geometry assumed in the analyses</td>
<td>Maintain geometry assumed in the analyses</td>
<td>Configuration control (A5.3.2.2, A5.6.2)</td>
<td>A</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Provide confinement for spent nuclear fuel</td>
<td>Maintain confinement during normal operations</td>
<td></td>
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<tr>
<td>Transportation cask</td>
<td>A3.4.2.1</td>
<td>Provide confinement for spent nuclear fuel</td>
<td>Withstand lateral impact of MHM crossing into load-in/load-out area and crashing into a cask-MCO suspended by the receiving crane over the cask receiving pit</td>
<td>Configuration control (A5.6.1)</td>
<td>A</td>
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<tr>
<td>(A4.3.8)</td>
<td></td>
<td></td>
<td>Withstand a 60-in. drop of cask-MCO onto the CSB concrete deck</td>
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<tr>
<td>MHM seismic</td>
<td>A3.4.2.1</td>
<td>Prevent damage to the operating deck</td>
<td>Restrain MHM from hitting the operating deck</td>
<td>Operability of various MHM design features (A5.5.1.1, A5.6.10)</td>
<td>A</td>
</tr>
<tr>
<td>restraint system (A4.3.9)</td>
<td></td>
<td></td>
<td></td>
<td>Visual verification of seismic restraints (pressure gauges and locking pins) (A5.5.3.3)</td>
<td></td>
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<tr>
<td>A4.13</td>
<td></td>
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</table>

Table A4-1. Summary of Safety-Class Systems, Structures, and Components. (4 sheets)
<table>
<thead>
<tr>
<th>Safety class SSC (Chapter A4.X section)</th>
<th>Accident section (Chapter A3.X section)</th>
<th>Safety function</th>
<th>Functional requirements</th>
<th>Performance criteria (TSR applicability, Chapter A5.X section)</th>
<th>ITS category</th>
</tr>
</thead>
<tbody>
<tr>
<td>MHM rails and rail frogs (A4.3.10)</td>
<td>A3.4.2.1</td>
<td>Prevent damage to the operating deck</td>
<td>Restrain MHM from hitting the operating deck</td>
<td>Periodic surveillance of rail clip hold-down bolt torque (A5.5.3.3)</td>
<td>A</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Ensure functionality of MHM seismic clamps and restraints</td>
<td></td>
<td>Periodic verification that rail frog hold-down bolts are snug (A5.5.3.3)</td>
<td></td>
</tr>
<tr>
<td>Item no.</td>
<td>Requirement area</td>
<td>Compliance summary</td>
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<tr>
<td>1</td>
<td>Fire protection</td>
<td>NRC equivalency is evaluated by the fire hazard analysis report (HNF-SD-SNF-FHA-002). The requirements of DOE Orders 5480.7A and 6430.1A provide adequate fire protection to achieve nuclear safety equivalency.</td>
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<tr>
<td>2</td>
<td>Natural phenomena</td>
<td>The SNF Project's seismic and other natural phenomena criteria achieves NRC equivalence in safety. Loss of shared utilities is evaluated to not be a safety concern for CSB.</td>
<td></td>
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<tr>
<td>3</td>
<td>Electrical equipment qualification</td>
<td>CSB design does not result in any &quot;harsh&quot; environment conditions. No active electrical systems are required during or after off-normal or postulated accident conditions.</td>
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<tr>
<td>4</td>
<td>Loss of AC power</td>
<td>Loss of AC power was evaluated by the CSB hazard analysis (HNF-SD-SNF-HIE-001) and this FSAR. The CSB has no safety-class electrical power loads and all systems fail safe on loss of AC power.</td>
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</tr>
<tr>
<td>5</td>
<td>Lead storage battery power</td>
<td>Not applicable to CSB.</td>
<td></td>
<td></td>
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</tr>
<tr>
<td>6</td>
<td>Lead storage battery power</td>
<td>Not applicable to CSB.</td>
<td></td>
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<td></td>
</tr>
<tr>
<td>7</td>
<td>I&amp;C system design (IEEE 603-1991)</td>
<td>Not applicable to CSB.</td>
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<td></td>
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</tr>
<tr>
<td>8</td>
<td>Nuclear criticality alarms per ANSI/ANSI-8.3</td>
<td>Not applicable to CSB. Criticality detection and alarms are not required per DOE Order 5480.24, paragraph 7.b.(3).</td>
<td></td>
<td></td>
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<tr>
<td>9</td>
<td>Human factors</td>
<td>Applicable NRC requirements and guidance were considered as part of the human factors evaluation and design performed for the CSB.</td>
<td></td>
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<tr>
<td>10</td>
<td>Code class determination</td>
<td>Applicable NRC guidance was considered (e.g., NRC Regulatory Guide 1.26). ASME Section III, Subsection NC (ASME 1995), and ANSI/AISC N690-94 were selected as the appropriate vessel and piping codes for CSB safety-class applications.</td>
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<tr>
<td>11</td>
<td>ASME Section III (ASME 1995) code cases</td>
<td>Regulatory Guides 1.84 and 1.85 are not applicable for CSB safety class mechanical components.</td>
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<tr>
<td>12</td>
<td>HEPA filtration design and testing</td>
<td>NRC equivalency is provided by application of ASME N509-1989 and ASME N510-1989 for HEPA filter design and testing.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>13</td>
<td>Light water reactor fuel handling systems and spent fuel storage facility design requirements</td>
<td>A detailed evaluation of CSB design compliance with the guidance of standards used for NRC licensed fuel handling and spent fuel storage facilities (ANSI/ANS 57.1 and 57.2) were performed in order to establish that NRC equivalency was achieved as appropriate.</td>
<td></td>
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</tr>
<tr>
<td>14</td>
<td>Lesson learned from NRC Generic Letters 88-14, 89-10, and 89-13.</td>
<td>Generic Letters 88-14, 89-10, and 89-13 are not applicable to CSB.</td>
<td></td>
<td></td>
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</tr>
<tr>
<td>Item no.</td>
<td>Requirement area</td>
<td>Compliance summary</td>
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<td>---------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------</td>
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<tr>
<td>15</td>
<td>Reporting of defects</td>
<td>Defect and noncompliance reporting is implemented for CSB in a fashion comparable to 10 CFR 21.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>16</td>
<td>Quality assurance program approval</td>
<td>DOE approval is required for quality assurance programs and program changes. The DOE process while not identical to NRC regulation is judged to achieve objectives equivalent to NRC regulations (i.e., 10 CFR 50.54(a)).</td>
<td></td>
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</tr>
<tr>
<td>17</td>
<td>Occurrence reporting</td>
<td>Occurrence reporting consistent with DOE O 232.1A is implemented for CSB during design and construction and is considered to be consistent with the comparable NRC regulation (i.e., 10 CFR 50.55(e)).</td>
<td></td>
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</tr>
<tr>
<td>18</td>
<td>Quality assurance program</td>
<td>CSB quality assurance program meets ASME NQA-1. The ASME NQA-1 standard is an acceptable method for meeting 10 CFR 50, Appendix B or 10 CFR 72, Subpart G.</td>
<td></td>
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<td></td>
</tr>
<tr>
<td>19</td>
<td>Counterfeit/defective equipment</td>
<td>The SNF Project monitors industry experience (including NRC notices, bulletins, and circulars) for counterfeit and defective equipment.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>20</td>
<td>Radiation controls</td>
<td>No high-radiation areas are anticipated in CSB and public dose criteria will not be exceeded for normal operation and anticipated occurrences.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>21</td>
<td>Radiation exposure</td>
<td>Estimated dose-equivalent exposures for CSB are well below both DOE and NRC criteria.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>22</td>
<td>Deleted (see item 29)</td>
<td>Deleted</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>23</td>
<td>ALARA</td>
<td>NRC Regulatory Guide 8.8 was used for CSB design and ALARA programs.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>24</td>
<td>Safety analysis report format and content</td>
<td>Inclusion of the appropriate format and content information is assured by use of the check list from HNF-SD-SNF-SP-012, Table 3.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>25</td>
<td>Effluent monitoring</td>
<td>Effluent monitoring is in compliance with 10 CFR 20 and 10 CFR 70.59. The 10 CFR 835 requirements are not applicable.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>26</td>
<td>General design criteria</td>
<td>Compliance with DOE Order 6430.1A addresses much of the 10 CFR 50, Appendix A general design criteria. The criteria specific to nuclear reactor systems are not applicable to CSB.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>27</td>
<td>Criticality safety value for k_{eff}</td>
<td>A criticality analysis of dry SNF contained in MCOs demonstrates that the $k_{eff}$ is well below 0.95 and a criticality at the CSB has been shown to be incredible.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>28</td>
<td>Fuel storage facility requirements</td>
<td>The design criteria for CSB equipment and facilities complies with the requirements contained in ANSI/ANS 57.9.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>29</td>
<td>Safety classification</td>
<td>CSB implementation of NRC &quot;important to safety&quot; classification is documented in this CSB FSAR.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Item no.</td>
<td>Requirement area</td>
<td>Compliance summary</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>---------</td>
<td>-----------------------------------------------------------------------------------</td>
<td>-----------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>24</td>
<td>HNF-SD-SNF-HIE-001, 2000, Canister Storage Building Hazard Analysis Report, Rev. 3, Fluor Hanford, Richland, Washington.</td>
<td>25</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
# Table A4-2. Summary of Compliance with Additional U.S. Nuclear Regulatory Commission Equivalency Items. (4 sheets)

<table>
<thead>
<tr>
<th>Item no.</th>
<th>Requirement area</th>
<th>Compliance summary</th>
</tr>
</thead>
<tbody>
<tr>
<td>5</td>
<td>NRC Regulatory Guide 8.8, 1982, <em>Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable</em>, U.S. Nuclear Regulatory Commission, Washington, D.C.</td>
<td></td>
</tr>
</tbody>
</table>

ALARA = as low as reasonably achievable.
CSB = Canister Storage Building.
DOE = U.S. Department of Energy.
FSAR = final safety analysis report.
HEPA = high-efficiency particulate air (filter).
I&C = instrumentation and control.
MCO = multi-canister overpack.
NRC = U.S. Nuclear Regulatory Commission.
SNF = spent nuclear fuel.
intake and exhaust plena. The subsurface structure is classified safety class in accordance with DOE Order 6430.1A because it is required to prevent failure of MCOs from the design criteria NPHs and DBA events described in Section A1.5 (Table A1-1) and Chapter A3.0 and to maintain the MCOs below the design fuel centerline temperature. The air intake and exhaust plena are integral features of natural convection cooling that prevent the violation of concrete design temperature criteria described in Section A4.3.1.3. The plena are structurally integral with the reinforced concrete below-grade exterior walls. The CSB subsurface structure is also classified ITS in accordance with 10 CFR 72, and Category A in accordance with the graded approach outlined in HNF-SD-SNF-DB-003.

A4.3.1.2 System Description. The CSB subsurface structure is described in Section A2.4.1 and shown on Figures A2-5, A2-6, and A2-7. The CSB subsurface is composed of reinforced concrete structures that divide the below-grade vault into three separate storage vaults and dedicated intake and exhaust plena. The design and construction of these structures provide protection from all applicable NPH and DBA events for the 220 storage tubes containing MCOs in vault 1 and for the 6 overpack tubes that may contain MCOs. These structures also are an integral part of the convective cooling system needed to maintain the MCOs below the design SNF centerline temperature, provided in HNF-SD-SNF-SARR-005, Multi-Canister Overpack Topical Report, and the concrete below ANSI/ACI 349-85 temperature limits for long-term normal operation. Vaults 2 and 3 are reserved for future missions. All three vaults provide means to anchor storage tube assemblies to embedded plates located in strips on the basemat directly below the operating deck embeds (see Section A4.3.2).

A4.3.1.3 Functional Requirements. The reinforced concrete subsurface structure is required to withstand all applicable NPH and DBA events to ensure that the MCOs and the storage tubes are not damaged. Maintaining the vault concrete temperature below 150 °F protects the vault’s concrete from long-term thermal degradation (HNF-SD-SNF-SARR-005) and maintains the MCO wall temperature below 270 °F. The air intake and exhaust plena are required to withstand all applicable NPHs to ensure that cooling is maintained to the vault and MCOs. The NPH events applicable to the CSB subsurface structure are discussed in Section A1.5. The NPH design requirements bound all DBAs identified in Chapter A3.0 for the reinforced concrete vault.

The design basis structural analysis of the CSB vault 1 demonstrates that the structure will be capable of withstanding the following load conditions:

- Reinforced concrete dead load
- Continuous live load on deck (100 lb/ft² plus tube vent and purge cart loads)
- Dead weight of the vault 1 exhaust stack
- Loadings associated with all three air-intake structures, steel and concrete portions (includes amplification of top steel canopy structure)
- Loadings associated with 220 standard and 6 overpack steel and concrete tube plugs (dead load supported by tube, seismic horizontal forces at deck, vertical at basemat)
- Loadings associated with 220 standard and 6 overpack steel storage tubes (dead load supported by tube base assemblies located in basemat embeds)
- Dead weight of 440 MCOs including impact absorbers
- Dead weight and live loads associated with MHM, 1,000,000 lb
- Dead weight and live loads associated with MCO cask transporter (live load in load-in/load-out trailer vestibule), 104,000 lb
- Dead weight and lifting loads associated with receiving crane (in load-in/load-out area), 330,000 lb
- Loading associated with the MCO sampling/weld area with seven pits and related equipment; two gantry cranes, sample cart, weld equipment, exhaust equipment
- Operating shelter structure weight (dead load, live load, tornado wind, snowfall and ashfall, seismic loads), 2,160,000 lb
- Side pressure on vault due to soil pressure (at rest) and surcharge load (200 lb/ft²)
- Thermal operating loads, based on Calculations CSB-S-0002, Vault Heat Transfer Analysis, and CSB-S-0003, Thermal Stress Analysis
- MCO concentric drop, 44 ft into storage tube, impact absorber (26 in. diameter)
- MCO eccentric drop, approximately 8 ft onto tube sealing flange, impact on deck
- Dead load of cask and MCO in cask receiving pit and associated gantry crane and tent enclosure in load-in/load-out area
- Loading associated with the Fast Flux Test Facility (FFTF) and MHM maintenance pits in load-in/load-out area
- DBE response spectral analysis with input motion at top of competent layer (8 ft below surface) and dynamic soil shear modulus variation of 1.5 and 1-over-1.5 times the best estimate value with confirmatory analysis using C_v = 1.0
- Tornado wind loads on stacks and intake structures that are transferred to the deck and vault.
A4.3.1.4 System Evaluation. The CSB below-grade structures are designed to withstand the natural phenomena specified in Section A1.5. The CSB basement foundation coincides with a soil profile identified as stratum 3 in Section 3.6.1.9 of WHC-SD-HWV-PSAR-001, Hanford Waste Vitrification Plant Preliminary Safety Analysis Report. This layer exhibits light cementation and is typically in a dense to very dense condition. The formation extends to a depth of approximately 250 ft below grade. Geotechnical investigations reported in WHC-SD-HWV-PSAR-001 characterized the soil properties at the elevation of interest as having high strength and low compressibility. Stratum 3 has a vertical static subgrade modulus value of 518 kip/ft$^2$ based on a 1 ft$^2$ plate. This value is adjusted for the actual foundation size by the equation in Report 10805-385-016, Report of Geotechnical Investigations for the Proposed Hanford Waste Vitrification Plant, Hanford Washington (Dames and Moore 1989).

The structural loading to the foundation, assuming all vaults are fully loaded with SNF, is well within the allowable soil bearing values. Details of the analysis of the adequacy of the soil to carry the anticipated foundation loading are documented in Calculation CSB-S-0023, HCSA/SNF Gravity Load Analysis (W/Tornado). The soil bearing pressures beneath the vault are less than 8 kip/ft$^2$, which is well within the allowable bearing capacity of 14 kip/ft$^2$. The factor of safety against sliding during a DBE is approximately 1.8 as determined in Calculation CSB-S-0024, Vault Seismic Load Analysis. Further soil compaction and settling and vault sliding analyses were performed because of the cantilever effect of the sampling/weld area. Transmittal FRT-011, Deck Cantilever Settlement and Vault Sliding Estimates (Bedell 1996b), concludes that only negligible settlement will occur in the compacted soils around the CSB and under the sampling/weld area and that vault wall pressures during a DBE are adequately considered.

HNF-PRO-097 provides criteria that apply to the design of the portion of the CSB requiring mitigation against NPH loads. Additional natural phenomena requirements associated with NRC equivalency, notably, tornado hazards, ashfall, and increased probable maximum precipitation amounts, are listed on Table A1-1. These loads take into account the seismic criteria of Item 2 of HNF-SD-SNF-DB-003. The facility design has been evaluated for applicable natural phenomena loads listed on Table A1-1 for the SSCs impacted. These loads are higher than loads in HNF-PRO-097, particularly those associated with tornado wind and probable maximum precipitation effects. For example, the tornado wind speed from Table A1-1 is given as 200 mi/h total, while the HNF-PRO-097 tornado wind speed is given as 175 mi/h for reactors. Higher wind speed on the superstructure transmits higher loads to the below-grade portion of the CSB. The design of the CSB takes into account the NPH loads from Section A1.5. The analysis of the subsurface structure design is contained in CSB-S-0042, Vault Design — Confirmation. A summary of the results (demand-to-capacity ratios) of structural analysis and design for the operating area deck, load-in/load-out area, sampling/weld station, and support area building foundation is provided in the structural analyses (Bedell 1996a). Table A4-3 summarizes the demand-to-capacity ratios and the governing load combinations from the structural analysis calculations. The demand-to-capacity ratios for many of the other calculated loads not shown in Table A4-3 are much lower than the demand-to-capacity ratios shown in this table. For all governing load combinations, structural demands are within structural capabilities.
<table>
<thead>
<tr>
<th>Region</th>
<th>Demand/capacity ratio</th>
<th>Governing load combination</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Vault Walls and Basemat</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>West wall Horizontal inside face reinforcing</td>
<td>0.98</td>
<td>9A: 1.05ED+L+H+Toh+Ed/Fu</td>
</tr>
<tr>
<td>West wall Horizontal outside face reinforcing</td>
<td>0.98</td>
<td>4A: 0.9FD+L+H+Toh+Ed/Fu</td>
</tr>
<tr>
<td>East wall Horizontal inside face reinforcing</td>
<td>0.82</td>
<td>2B: 1.05*1.05FD+1.3L+1.3H+1.05Toh</td>
</tr>
<tr>
<td>North wall Vertical south face reinforcing</td>
<td>0.95</td>
<td>11B: 0.9ED+L+H+Toh+Ed/Fu</td>
</tr>
<tr>
<td>Interior north wall Vertical north face reinforcing</td>
<td>0.95</td>
<td>9A: 1.05ED+L+H+Toh+Ed/Fu</td>
</tr>
<tr>
<td>South wall Vertical north face reinforcing</td>
<td>0.90</td>
<td>4B: 0.9FD+L+H+Toh+Ed/Fu</td>
</tr>
<tr>
<td>Basemat E/W top reinforcing</td>
<td>0.70</td>
<td>3D: 1.05FD+L+H+Toh+Ed/Fu</td>
</tr>
<tr>
<td><strong>Vault Operating Floor</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Operating floor at tubes E/W top reinforcing</td>
<td>0.88</td>
<td>1.05ED + L + H + Toh + Ed/Fu</td>
</tr>
<tr>
<td>Operating floor at crane rails N/S flexural shear</td>
<td>0.99</td>
<td>1.05ED + L + H + Toh + Ed/Fu</td>
</tr>
<tr>
<td>Operating floor at intake plenum E/W top reinforcing</td>
<td>0.99</td>
<td>1.05FD + L + H + Toh - Ed/Fu</td>
</tr>
<tr>
<td>Operating floor at exhaust plenum N/S top reinforcing</td>
<td>0.99</td>
<td>1.05ED + L + H + Toh - Ed/Fu</td>
</tr>
<tr>
<td><strong>Sampling/Weld Area</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Walls Inside vertical reinforcing</td>
<td>0.97</td>
<td>10D: 1.05ED+Ed/Fu</td>
</tr>
<tr>
<td>Pits basement E/W bottom reinforcing</td>
<td>0.70</td>
<td>3A: 1.05FD+L+To+H+Ed/Fu</td>
</tr>
<tr>
<td>South cantilever deck N/S top reinforcing</td>
<td>0.96</td>
<td>9B: 0.9ED+L+H-Ed/Fu</td>
</tr>
<tr>
<td><strong>Load-In/Load-Out Area</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Walls Flexural shear with reinforcing (out of plane)</td>
<td>0.95</td>
<td>1.05ED+L+H+Ed</td>
</tr>
<tr>
<td>Basemat Flexural reinforcing</td>
<td>0.82</td>
<td>1.05ED+L+H+Ed</td>
</tr>
<tr>
<td><strong>MHM Service Pit</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Walls Flexural reinforcing</td>
<td>0.96</td>
<td>3A: 1.05FD+Ed/Fu</td>
</tr>
<tr>
<td>Basemat E/W flexural shear (out of plane)</td>
<td>0.90</td>
<td>10A: 0.9ED+Ed/Fu</td>
</tr>
<tr>
<td><strong>Cask Receiving Pit</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Walls Inside vertical reinforcing</td>
<td>0.97</td>
<td>10D: 1.05ED+Ed/Fu</td>
</tr>
<tr>
<td>Basemat E/W bottom reinforcing</td>
<td>0.67</td>
<td>3A: 1.05FD+L+To+H+Ed/Fu</td>
</tr>
<tr>
<td>North cantilever deck N/S top reinforcing</td>
<td>0.96</td>
<td>7B: 1.05ED+L+H-Ed/Fu</td>
</tr>
</tbody>
</table>
Table A4-3. Summary of Demand-to-Capacity Ratios and Governing Load Combinations. (3 sheets)

<table>
<thead>
<tr>
<th>Region</th>
<th>Demand/capacity ratio</th>
<th>Governing load combination</th>
</tr>
</thead>
<tbody>
<tr>
<td>Exhaust Stack Pedestals (Calculation CSB-S-0045)*</td>
<td>0.99</td>
<td>3B: 1.05FD+L+H+Tol+Ed/Fu</td>
</tr>
<tr>
<td>Walls</td>
<td>Horizontal inside reinforcing</td>
<td></td>
</tr>
<tr>
<td>Intake Concrete Towers (Calculation CSB-S-0046)*</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Vault 1 tower walls†</td>
<td>Vertical flexural shear</td>
<td>0.84</td>
</tr>
<tr>
<td>Structural element</td>
<td>Demand/capacity ratio</td>
<td>Governing load combination</td>
</tr>
<tr>
<td>Intake Steel Structure (Calculations CSB-S-0015, CSB-S-0028, and CSB-S-0047)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Anchor bolts</td>
<td>0.86</td>
<td>0.9D+Ed/Fu</td>
</tr>
<tr>
<td>Shear plate at base</td>
<td>0.22</td>
<td>D+Ed/Fu</td>
</tr>
<tr>
<td>Roof plate flexure</td>
<td>0.30</td>
<td>D+S+Wt</td>
</tr>
<tr>
<td>Columns</td>
<td>0.27</td>
<td>D+S+Wt</td>
</tr>
<tr>
<td>Braces</td>
<td>0.66</td>
<td>D+Ed</td>
</tr>
<tr>
<td>Exhaust Stack for Vault 1 (Steel) (Calculation CSB-S-0012)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Stack flexure at base</td>
<td>0.83</td>
<td>1.05D+L+Wt</td>
</tr>
<tr>
<td>Anchor bolts</td>
<td>0.82</td>
<td>0.9D+Wt</td>
</tr>
<tr>
<td>Support Building (Calculation CSB-S-0019)</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Frame girders</td>
<td>Combined stress</td>
<td>0.69</td>
</tr>
<tr>
<td>Columns</td>
<td>Combined stress</td>
<td>0.71</td>
</tr>
<tr>
<td>Exhaust stack</td>
<td>Anchor bolts</td>
<td>0.92</td>
</tr>
<tr>
<td>Operating Building (Calculation CSB-S-0048)*</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Structural element</td>
<td>Element no.</td>
<td>Demand/capacity ratio</td>
</tr>
<tr>
<td>Columns W30X235 W/2MC18X42.7</td>
<td>1553</td>
<td>0.76 T</td>
</tr>
<tr>
<td>Columns W30X235</td>
<td>1452</td>
<td>0.83 C</td>
</tr>
<tr>
<td>Truss top chord W14X90</td>
<td>1570</td>
<td>0.66 C</td>
</tr>
<tr>
<td>Truss bottom chord W14X61</td>
<td>1560</td>
<td>0.92 C</td>
</tr>
<tr>
<td>Truss web WT8X33.5</td>
<td>1580</td>
<td>0.97 C</td>
</tr>
<tr>
<td>Truss diagonal WT8X33.5</td>
<td>1389</td>
<td>0.86 C</td>
</tr>
<tr>
<td>Vertical bracing WT8X33.5</td>
<td>1713</td>
<td>0.96 C</td>
</tr>
<tr>
<td>Horizontal bracing WT7X21.5</td>
<td>1827</td>
<td>0.81 C</td>
</tr>
<tr>
<td>Column anchor bolts (8-1 in. diameter)</td>
<td></td>
<td>0.85</td>
</tr>
</tbody>
</table>
## Table A4-3. Summary of Demand-to-Capacity Ratios and Governing Load Combinations. (3 sheets)

<table>
<thead>
<tr>
<th>Structural element</th>
<th>Demand/capacity ratio</th>
<th>Governing load combination</th>
</tr>
</thead>
<tbody>
<tr>
<td>Vertical stiffeners</td>
<td>Weld</td>
<td>0.68</td>
</tr>
<tr>
<td>Embed weld studs</td>
<td>Shear</td>
<td>0.76</td>
</tr>
<tr>
<td>Basement</td>
<td>Punching shear</td>
<td>0.31</td>
</tr>
</tbody>
</table>

*The design basis extreme temperature maximum and minimum are assumed to occur for a duration of 6 hours.

To = Normal operating temperature.

Toh = Internal forces as a result of normal high operating temperature. They are assumed equal to the forces determined for the extreme high ambient temperature of 115°F.

Tol = Internal forces as a result of normal low operating temperature. They are assumed equal to the forces determined for the extreme low ambient temperature of -27°F.

The thermal loads are negligible since the ambient temperature inside and outside is same.

*Only Vault 1 tower will be constructed for the Spent Nuclear Fuel project.

'1. Assumes two MCOs at 18,100 lb each.

2. Tube assembly weight is 5,450 lb.

3. Assumes two impact absorbers at 500 lb each.

FFT = Fast Flux Test Facility.

MCO = multi-canister overpack.

MHM = multi-canister overpack handling machine.

C = Compression.

D = Dead load.

Ed = Design basis earthquake.

ED = Dead load, vaults empty with MCOs.

FD = Dead load, vaults full with MCOs.

Fu = Inelastic energy absorption capacity factor.

H = At-rest soil pressure load.

L = Live load (floor, roof, snow, ashfall).

S = Snow load.

T = Tension.

Wt = Tornado wind load.

The following referenced calculations were prepared by Fluor Daniel, Incorporated, Richland, Washington:

- CSB-S-0042, 1996, *Vault Design — Confirmation, Rev. 0*.
- CSB-S-0043, 1996, *HCSA Deck Design — Confirmation, Rev. 0*.
- CSB-S-0046, 1997, *Air-Intake Concrete Structure — Confirmation, Rev. 0*.
- CSB-S-0047, 1997, *Air-Intake Steel Structure — Confirmation, Rev. 0*.
The subsurface concrete structure forms the major portion of the natural convection cooling design for the CSB. Design of the natural convection cooling features is based upon two-dimensional thermal analysis using two-dimensional modeling software and assuming the maximum design air inlet temperature of 115 °F. Results of the vault thermal analysis are documented in Calculation CSB-HV-0001, *A Canister Storage Building Vault Thermal Analysis*. The results indicate that the maximum air temperature inside the vault is less than the 150 °F concrete limitation in ANSI/ACI 349-85, Appendix A. Figures A4-2 and A4-3 represent the results of the thermal analyses and should be referred to for temperature profiles. The results of calculation CSB-S-0042 are based on the assumption that the design basis extreme temperature maximum occurs for a duration of 6 hours. For conservatism, the reinforced concrete strength values used in the analysis were decreased by 1,000 lb/in² to account for long term concrete temperature effects. The specified reinforced concrete 90-day compressive strength (Class MC1P) for the subsurface concrete structures (below elevation 704 ft) is 5,000 lb/in² in accordance with ANSI/ACI 349-85. Steel reinforcement is specified as 60,000 lb/in² tensile strength in accordance with ASTM A615/A615M-96a, *Standard Specification for Deformed and Plain Billet-Steel Bars for Concrete Reinforcement*. For shielding purposes, the density of the subsurface concrete is assumed to be 2.24 g/cm³.

Detailed design calculations have not addressed solar heat gain and wind effects because those effects are of minor importance. The underground storage vault is not directly affected by solar heat. The exhaust stack and the air intake are exposed to the same incidence of instantaneous solar exposure, which has a negligible effect on the thermal analysis because of the slight increase in density differential created by the stack effect. The thermal analysis does not include this effect and has considered the effect to be a part of the safety factor in the calculation. The above-grade structures that rest on the below-grade intake and exhaust plena are designed to prevent plugging caused by ice, snow, ashfall, and windborne debris. Evaluations of flow blockage contained in CSB-HV-0003, *Spent Nuclear Fuel Vault Loss of Cooling Analysis* (see Section A3.4.2.6), demonstrate that the increased temperatures due to blockage do not challenge the structural integrity of the CSB vault.

**A4.3.1.4.1 Soil-Structure Interaction.** Several steps were used to determine the effects of soil-structure interaction (SSI) on the seismic response of the SNF CSB. Initially, parametric studies were conducted to evaluate the seismic response of the site soils and to determine the level of the soil stress-strain properties for the given level of input ground motion. This was accomplished by finite element methods using the SHAKE computer code analysis. One-dimensional models of the soil properties were developed and equivalent linear soil properties compatible with the strains developed in each layer. These properties were then used in the next step as an initial estimate of soil properties used in the SSI analysis. The SSI was defined by finite element methods using the FLUSH computer code analysis, which is described in *FLUSH: A Computer Program for Approximate 3-D Analysis of Soil–Structure Interaction Problems* (Lysmer et al. 1975).

Two-dimensional models of the soil and structure were developed and gross motion SSI responses for the structure and loads in the below-grade walls and basemat were assessed.
These results served as verification of the three-dimensional dynamic model created using the
SAP90 computer program documented in *SAP90, A Series of Computer Programs for Finite
Element Analysis of Structures* (CSI 1995). The model included the SNF CSB underground vault
with surrounding soil and a massive soil column representing "free field." The outer perimeter
nodes of the soil elements surrounding the vault were slaved to the elements of the massive soil
column thus simulating the transmitting boundary as in the FLUSH (Lysmer et al. 1975) analysis.
This forced the outer perimeter boundaries of the soil surrounding the vault to move with the free
field. This analysis approach provided a one-step method to account for SSI effects and yielded
three-dimensional shear and moment force distributions over the underground vault structure.
The three-dimensional analysis used input (response spectra and time histories) at the bottom of
the vault generated from the free field ground motions acting at the top of the competent layer,
deconvoluted to the bottom of the vault using the SHAKE program.

Uncertainty in the accurate prediction of dynamic soil properties was accounted for by
employing a parametric analysis using a range of soil dynamic shear moduli values. The range of
soil dynamic shear moduli was bounded by the best estimate dynamic shear modulus multiplied by
1.5, the best estimate dynamic modulus divided by 1.5, and the best estimate shear modulus
multiplied by 1.0 (ASCE-4-86). Confirmatory analyses for soil dynamic shear moduli ranging
from 2 and 0.5 times the best estimate were also done. Strain-dependent soil shear modulus and
damping values were used in the SSI analysis.

The SAP90 computer program (CSI 1995) is not normally used to account for SSI.
However, applying the boundaries of soil elements, using the appropriate properties and applying
the free-field motion for the surface and below-surface soil pressures, the resultant response
spectra were shown to be consistent with the FLUSH (Lysmer et al. 1975) analysis values. It is
on this basis that the SAP90 computer model is reasonable and the analysis results are
appropriate. The comparison results were documented in Letter FRF-092, *DOE SAR Team

A4.3.1.4.2 Time Histories. ABB Impell Corporation was a subcontractor to the
architect-engineer on the Hanford Waste Vitrification Plant Project. They provided the artificial
earthquake time history that was used in the design of the Hanford Waste Vitrification Plant CSB
and is being used in the SNF CSB Project. *Seismic Acceleration Time Histories in Support of
Fluor Daniel, Inc. Project HWVP-845734* (Salmon et al. 1990) indicates that a single synthetic
time history was created to simulate the DBE. The resulting time history is shown to meet NRC
acceptance criteria for a single design time history (NUREG-0800). All important ground motion
characteristics, including check for power spectral density content, were included in the developed
time history.

A4.3.1.4.3 Asymmetric Loading Effects on the Substructure. The seismic analysis
considered all three vaults to be filled with MCOs and all three intake structures and exhaust
stacks to be in place. In addition to all the actual above-grade eccentricities (e.g., MHM,
superstructure, and receiving crane), the model included a 5% accidental eccentricity.
A dominant feature of these below-grade vaults is the participation of the soil mass in the wall
flexure and in-plane shears. Since the stored SNF, intake structures, and exhaust stacks account
for less than 1% of the total seismic model mass (soil, structure, and stored fuel) and
approximately 32% of the vault structure and stored fuel system, it was judged that the effective
eccentricity was small and that the governing configuration for seismic analysis would be the
condition in which all three vaults are full and all three intake and exhaust structures in place.
A two-dimensional, 1-ft strip model of the vault-deck section was evaluated for the asymmetric
gravity and temperature loads. The results of the analysis confirmed that the demand in moments
and shears did not exceed the capacity (Jacobs 1996).

The design of the basemat and below-grade walls is in accordance with ANSI/ACI 349-85.
In general the governing design condition for the concrete deck, below-grade walls, and basemat
was based on thermal and seismic rather than tornado loads. The extreme high-temperature
condition only occurs when the vault is completely full and the extreme low-temperature
condition occurs only when the vault is completely empty. Any asymmetric loading of the vault
(e.g., all tubes in one or two vaults full of heat-source material, but not all three vaults loaded)
relaxes the self-straining thermal stresses in the vault structure and reduces thermal demand.
Based on this reasoning, it was judged that the governing condition for the design of the basemat
and below-grade walls was either all the vaults full with high temperature inlet air or all vaults
empty with low temperature inlet air. Therefore, the designs of the basemat and below-grade
walls are bounded by the two extreme conditions and are not governed by a checkerboard loading
pattern because of the dominance of thermal loads.

The internal forces due to temperature used in the design are based on the design basis
extreme ambient temperature of 115 °F maximum and -27 °F minimum (Bedell 1996a) occurring
for a duration of 6 hours at any time during the service life of the CSB based on site
meteorological data. Therefore the demand-capacity ratios with the temperature load $T_a$ include
conservatism as loads caused by extreme temperature conditions have been used in lieu of normal
operating temperatures as required by ANSI/ACI 349-85. The demand-capacity ratios given are
for seismic confirmatory analysis done with seismic input at the top of the competent layer (8 ft
below the surface) and using soil shear modulus variations between best estimate values multiplied
by $(1 + C_v)$ and best estimate values divided by $(1 + C_v)$, where $C_v$ is a factor that accounts for
uncertainties in the SSI analysis and soil properties. The minimum value of $C_v$ according to
ASCE-4-86 is 0.5; however, for confirmatory analysis, $C_v = 1.0$ was used.

The high mass of the generally box-like configuration formed by the below-grade concrete
CSB structure, the deck, and the above-grade concrete features (the concrete portion of the inlet
structure and the exhaust stack bases) makes it susceptible to DBE forces. When combined with
thermal stresses, governing load combinations result for many of these components.

The deck has been designed to provide an adequate safety margin for combined thermal
and tornado wind loading to prevent a 200 mi/h tornado wind load from creating unacceptable
loads on areas where the bending moment is transmitted from the above-grade structures to the
deck. Tornado wind loading by itself is governing for some features of the operating area shelter
as well as for the support building.
A4.3.1.4.4 Modeling of the Deck, Superstructure, Ventilation Stacks, and the Multi-Canister Overpack Handling Machine — Effects on the Substructure. As severe as tornado loading may be, the forces transmitted to the below-grade structure are well below the forces transmitted by seismic forces. Calculations indicate that the below-grade design controlling values of soil pressures, member stresses, displacements, or other parameters are not affected by tornado wind (Wt) loads. None of the governing conditions for below-grade items (the substructure) were based on tornado wind loads. The weight of the MHM is included in the seismic calculation that affects the seismic response of the CSB below-grade and at-grade structures. The design also includes evaluations of the deck and related structures, such as the stacks and inlets for tornado wind effects as well as the 5% dead load increases required by ANSI/ANS-57.9-1992 and Item 28 of HNF-SD-SNF-DB-003. The designs were modified as appropriate to accommodate those loads before releases for construction of the below-grade and at-grade reinforced concrete structures.

A4.3.1.4.5 Vault Sliding Resistance and Passive Wall Pressures. Analysis to determine the level of strain required to slide (see Bedell 1996a), as compared with calculated strain values, was performed to demonstrate that relevant passive wall pressures were used in the design of the below-grade walls (see CSB-S-0024). The analysis contained in the Design Basis Document (Bedell 1996a) estimates that full frictional resistance of the vault basemat-soil interface occurs at a displacement of only 0.15 in., whereas full passive soil pressures on the wall are activated at a movement of approximately 14 in. for a 47-ft-deep vault wall.

The FLUSH analysis documented in CSB-S-0032, FLUSH Analysis — MEAN 1.5 and MEAN/1.5, indicates that the maximum shear strain and shear modulus in the soil elements below the vault basemat result in a DBE shear stress at the soil basemat interface of 1.54 kip/ft². Obtaining the frictional resistance of the mass concrete on coarse sand and gravel (2.38 kip/ft²) results in a factor of safety against sliding of 1.55. The 0.15-in. displacement for full frictional resistance to be activated and the ratio of shear stress to frictional resistance of 0.65 results in a displacement of 0.10 in. due to DBE shear. For shear deformation of 0.15 in. or less, the passive pressure coefficient is given as 0.50 (Bedell 1996a). The CSB walls were designed for a passive pressure coefficient of 0.62, which is greater than the passive pressure occurring from vault displacement during a DBE. In summary, the CSB vault walls have been designed for larger soil pressures than the passive pressures developed during a DBE.

A4.3.1.5 Controls (Technical Safety Requirements). The CSB subsurface structure is a design feature (see Section A5.6), and its safety function is managed by the configuration control process.

A4.3.2 Carbon Steel Basemat Embeds

A4.3.2.1 Safety Function. The 220 tubes (and 6 overpack tubes) of vault 1 are seismically restrained through the use of tube base assemblies that bolt to carbon steel embeds in the basemat of the CSB. In vault 1 these embeds are safety class in accordance with DOE Order 6430.1A and
ITS Category A in accordance with the graded approach in HNF-SD-SNF-DB-003 because they serve the safety function to maintain horizontal stability of standard and overpack storage tubes, provide for natural convective cooling, and prevent damage to MCOs caused by structural failure.

A4.3.2.2 System Description. Two rows of carbon steel embed strips are located under each row of tubes, as shown in Figure A2-5. The embeds serve as the welding plates for the threaded weld studs to which the tube base assemblies are bolted. The tube base assemblies are not affixed to the tubes but will restrain the tubes. Each tube base assembly comprises a 1-in. carbon steel plate bolted to the embed strips in four places and a short (about 6 in. tall), right-circular cylinder of 0.5-in.-thick carbon steel, of a diameter slightly larger than that of the tube, welded to the top of the plate. The tubes rest on top of the carbon steel plates and are seismically restrained within the 6-in.-tall cylindrical band.

A4.3.2.3 Functional Requirements. The embeds in vault 1 must remain in place during the DBE to provide horizontal stabilization of the tubes. Horizontal stability of the storage tubes is required during the DBE to maintain cooling.

A4.3.2.4 System Evaluation. The carbon steel embeds for vault 1 have been evaluated for the DBE specified in Section A1.5 based on the seismic loading from the storage tubes, impact absorbers, and the MCOs, which contain SNF and weigh up to 20,200 lb. Embed–tube interface design accommodates the DBE for tubes loaded with MCOs and impact absorbers. The embeds and tube base assemblies work in concert to restrain the storage tubes at the basemat elevation. Seismic analysis and design of the embeds was performed to ANSI/ACI 349-85 standards in calculation CSB-S-0007A, Storage Tube Analysis, and included the seismic interaction loads from two MCOs.

A4.3.2.5 Controls (Technical Safety Requirements). The carbon steel basemat embeds are a design feature (see Section A5.6), and their safety function is managed by the configuration control process. Design features for the carbon steel basemat embeds ensure that their safety function can be relied upon.

A4.3.3 Canister Storage Building At-Grade Structures

A4.3.3.1 Safety Function. The at-grade reinforced concrete structures of the operating area deck (including the sampling/weld area, the load-in/load-out area, and the intake structure and exhaust stack bases) are an integral part of the CSB vault and are classified as safety class in accordance with DOE Order 6430.1A. The CSB at-grade structures share the design functions of the vault to prevent damage to MCOs caused by structural failure and provide for natural convective cooling. The CSB at-grade structures are classified ITS in accordance with 10 CFR 72, and Category A in accordance with the graded approach outlined in HNF-SD-SNF-DB-003.
A4.3.3.2 System Description. The at-grade reinforced concrete structures are shown on Figures A2-3 and A2-7. The at-grade reinforced concrete structures have overall dimensions of 285 ft, 0.5 in. by 189 ft, 5.5 in. The operating area deck is a 5-ft-thick reinforced concrete structure. The deck is 181 ft, 2 in. by 165 ft, 9 in. Deck embeds in the operating area deck locate the openings for the 220 standard and 6 overpack storage tube locations. The deck embeds for vault 1 are fitted with tube plugs and covers designed to provide shielding and to provide access to service connections on the tube plugs (see Section A4.4.2). The deck embeds for vaults 2 and 3 are fitted with embed covers to protect personnel from falls (see Section A2.4.3.3). The deck embed sleeves in the operating area deck provide a finished surface on which to rest the storage tube bellows assembly (see Sections A4.3.4 and A4.3.5).

A 4 ft, 4 in. wide by 3 ft high curb borders the operating floor area (operating deck, sampling/weld area, and load-in/load-out area) on the east, west, and south. The curb height is increased to 8 ft at the north end to meet shielding requirements for the transportation cask while allowing unlimited occupancy in the support area building.

The intake structure and exhaust stack are described in Section A2.4.4.1.

The sampling/weld area is an extension to the south of the operating deck with overall dimensions of 138 ft, 11 in. by 35 ft, 3 in. A loading–staging area is provided at the west side of the sampling/weld area. The loading–staging area dimensions are 27 ft, 4 in. by 24 ft, 11 in.

The load-in/load-out area is located at the north end of the operating deck. The dimensions of the load-in/load-out area are 189 ft, 5.5 in. by 42 ft, 7 in. at the trailer vestibule and 189 ft, 5.5 in. by 29 ft, 9.5 in. at the MCO service station.

The at-grade structure of the CSB is designed to carry the loads associated with the MHM transporting an MCO. The MHM, a large, heavy crane, weighs approximately 990,000 lb, and the heaviest MCO weighs approximately 20,200 lb. The MHM rides on steel rails running north and south and embedded in the operating deck concrete. Anchor bolts and lateral restraints secure the rails within the trenches and are designed to provide restraint of the MHM rails, including the MHM bound to the rails by the MHM bridge seismic clamps, during a DBE (see Section A4.3.10 for a discussion of the MHM rails).

A4.3.3.3 Functional Requirements. The operating deck is designed to move with the vault walls and basemat to support the loads associated with the CSB. The operating deck, support area building foundation, load-in/load-out area, and sampling/weld area are designed to the same safety-class criteria and NPH events as the below-grade vault. The NPH events are those discussed in Section A1.5. The NPH design requirements bound all identified DBAs.
To meet these requirements, the operating area deck (including the sampling/weld areas and load-in/load-out area) is designed for the following loading conditions:

- Roof dead load (17 lb/ft² used in design, 15.44 lb/ft² estimated including a 5% contingency [0.74 lb/ft²] per ANSI/ANS 57.9-1992)
- Snow load at ground level; roof snow load (considering drifts at low roofs)
- Roof live load (20 lb/ft²)
- Volcanic ashfall load at ground level and ashfall roof drift
- Siding dead load
- Wind loads (90 mi/h; wind-driven missile 2 in. by 4 in. wood plank, 15 lb at 50 mi/h)
- Seismic load, horizontal response spectra anchored at 0.35 g (“free field” used in Calculation CSB-S-0029, CSB Operating Shelter); vertical is two-thirds of horizontal spectra; confirmatory analysis Calculation CSB-S-0048, Operating Shelter Confirmation Analysis, used in-structure response spectra at deck level based on seismic input at top of competent layer and soil properties at mean, two times mean, and one-half mean

A4.3.3.4 System Evaluation. The operating area deck has been designed to provide an adequate safety margin for combined thermal and tornado wind loading to prevent a 200 mi/h tornado wind load from creating unacceptable loads on areas where the bending moment is transmitted from the above-grade structures to the deck. Tornado wind loading by itself is governing for some features of the operating area shelter. The CSB at-grade reinforced concrete structures have been designed for all applicable loads and load combinations required by Table A1-1. These are further defined in the architect-engineer’s Design Basis Document (Bedell 1996a), Section 10. The loads include normal loads, severe and extreme environmental loads, and natural phenomena loads. The NPH loads bound all identified DBAs. A summary of the results (demand-to-capacity ratios) of structural analysis and design for the operating area deck, load-in/load-out area, sampling/weld station, and support area building foundation is provided in the structural analyses (Bedell 1996a). For all governing load combinations, structural demands are within structural capabilities.

The operating area deck has been designed using an isotropic plate model with a reduced modulus of elasticity derived from a comparison of the natural frequencies of a beam (grillage) model and a plate-element model. Demand capacity ratios derived from Calculation CSB-S-0041, SNF Deck Design and Exhaust Stack — Confirmation, indicate that the governing load combinations are those of dead and live load with a soil pressure load in combination with
extreme temperatures ($T_{el}$ and $T_{oh}$), and seismic loads (modified using $F_u > 1.0$ per
DOE-STD-1020-94). Demand/capacity ratios in excess of 90% during these conditions are
present in the operating floor at the crane rails (N–S flexural shear; 0.99), at the intake plenum
(E–W flexural shear; 0.98, E–W top reinforcing; 0.99, and N–S bottom reinforcing; 0.99), and at
the exhaust plenum (E–W flexural shear; 0.95, E–W bottom reinforcing; 0.95, and N–S top
reinforcing; 0.99). Highest demand/capacity ratios for the operating floor at the tubes were at the
E–W top reinforcing; 0.88, and E–W bottom reinforcing; 0.86 (Table A4-3).

Structural concrete mix classes for the operating area deck structures (above elevation
704 ft, 0 in.) have a specified 90-day compressive strength of 6,000 lb/in² (Class MC2P) in
accordance with ANSI/ACI 349-85. Those structures that are slab-on-grade (north and south
cantilevers), i.e., the load-in/load-out area, support building foundation and sampling/weld station,
have a specified 28-day compressive strength of 4,000 lb/in² in accordance with
ANSI/ACI 349-85. For conservatism, the reinforced concrete strength values of the operating
area deck structures used in the analysis were decreased by 1,000 lb/in² to account for long term
concrete temperature effects. Steel reinforcement for all at-grade structures is specified as
60,000 lb/in² tensile strength in accordance with ASTM A615/A615M-96a. For shielding
purposes the density of the operating area deck concrete is assumed to be 2.24 g/cm³.

The operating area deck has been designed to perform at the temperatures expected from
the below-grade storage of MCOs. The design analysis results, as shown on Figure A4-3,
indicate that for a fully loaded vault, the highest calculated concrete temperature on the operating
deck floor surface is 105 °F, coincident with a site ambient temperature of 115 °F. The results of
calculation CSB-S-0041 are based on the assumption that the design basis extreme temperature
maximum occurs for a duration of 6 hours. The highest calculated temperature for the operating
area deck slab on the vault side is 121 °F except for slightly higher temperatures in the local areas
adjacent to the storage tubes. This temperature occurs at the underside of the deck slab next to
the storage tube. These temperature values were obtained from CSB-HV-0001.

The geotechnical predicted-settlement values (Bedell 1996a) were compared with the
settlement values computed by the structural analysis for the sampling/weld station, which is
documented in Calculation CSB-S-0026A, HCSA/SNF Deck Design. The differential settlements
computed by the structural analysis were found to be generally larger than in the geotechnical
analysis. The maximum anticipated differential settlement is on the order of 0.10 in. across the
30 to 35 ft of the south cantilevered deck. Although the predicted differential settlements are
small, a crack control joint at the interface with the vault's north and south walls has been included
in the design.

The quality of backfill material being used and the field compaction tests conducted to date
have been reviewed and found to meet or exceed the project specification requirements
(Bedell 1996a). To account for uncertainty in the execution of construction and to confirm the
design agent's predicted settlement assumptions, settlement measurements using the survey
markers shown in Figure A2-3, sheet 1, at the north and south vault walls and at the edges of the
north and south deck cantilevers will be taken and evaluated after construction and during
operation.

Specific tornado wind loads were developed for the intake and exhaust stacks to evaluate
their impact on the vault deck. Structural evaluations are documented in Calculation
CSB-S-0012, Vault Exhaust Stack, for the exhaust stack (including exhaust stack base);
Calculations CSB-S-0028, Intake Structure, Steel (Tornado Loading); and CSB-S-0028A, Intake
Structure, Concrete (W/Tornado), for the intake structure; and Calculation CSB-S-0026, Deck
Design, for the operating area deck (CSB-S-0026A for the sampling/weld area). The change
from hot conditioning pits to sampling/weld stations is documented in Calculation CSB-S-0043,
HCSA Deck Design — Confirmation. The governing load combination with tornado wind for the
operating area floor is at the exhaust plenum (E–W top reinforcing, 0.42).

The columns of the operating area shelter are pin-connected to base plates located along
the curbs. Anchor bolts for these base plates are in accordance with ANSI/ACI 349-85. Moment
loadings associated with the partial fixity of these columns are carried by these curbs into the
operating deck and subsurface structure. Evaluations of the partial plate fixity condition are
included in CSB-S-0048, Supplement A, and determined that the anchor bolts can accommodate
moments varying between 20% for the 8-bolt pattern and 40-50% for the 12-bolt pattern, of the
fully fixed moments. These capacities are based on the 100-40-40 rule (as permitted by
DOE-STD-1020-94 for seismic component loadings) and also considered 17 lb/ft² dead load and
5 lb/ft² for the snow as additional effective mass for seismic analysis. The capability of the
operating deck and subsurface structures to withstand the added base plate fixity was evaluated
(see CSB-S-0041), and it was determined that the deck has the necessary capacity to carry the
added moment fixity load (Table A4-3).

Using NRC-accepted tornado missile probability methodology provided by the design
authority in Letter 9652531, Spent Nuclear Fuel/Canister Storage Building — Tornado Strategy
Implementation Plan (Bazinet 1996), the architect-engineer completed a probabilistic risk
analysis of tornado wind-driven missiles becoming credible impact loads for the CSB assuming no
resistance provided by the superstructure. Further adjustments to the probabilistic risk analyses
were made using a survey of the site for potential missiles and documented in
Letter 96-0133.MMB, Task Order 07, WHC Contract No. MRV-SBW-482901. Analysis of
Tornado/Tornado Missile Risk to New SNFP Facility; Initial Report (Beary 1996b), and in Letter
075MMB.96, Errata: Table 4. Analysis of Tornado/Tornado Missile Risk to New SNFP Facility
(Beary 1996a). The revised estimated annual frequency of 4 × 10⁻⁶ for a missile strike on the CSB
is well below the NRC criterion of 1 × 10⁻⁶ per year (Jacobs 1996). Therefore, the CSB is not
subject to missile impacts, and the structures are designed for the tornado wind and atmospheric
pressure change.

A4.3.3.5 Controls (Technical Safety Requirements). The CSB at-grade structures are design
features (see Section A5.6), and their safety functions are managed by the configuration control
process. Design features of the CSB at-grade structures ensure that their safety function can be
relied upon.
A4.3.4 Standard Storage Tubes and Tube Base Assemblies

A4.3.4.1 Safety Function. The standard storage tubes and tube base assemblies are classified as safety class in accordance with DOE Order 6430.1A and ITS Category A, using the graded approach of HNF-SD-SNF-DB-003. These components perform the safety functions of preventing damage to MCOs caused by structural failures and providing for natural convective cooling. These components are designated safety class and ITS Category A as derived in Section A3.4.2.1 and Section A3.4.4.1. Safety features of the storage tube design limit potential damage to an MCO and ensure that an MCO is “readily retrievable” before and after an accidental drop. This meets the NRC equivalency requirement in Item 2 of HNF-SD-SNF-DB-003 that the MCO be readily retrievable.

A4.3.4.2 System Description. The standard storage tubes are 0.5-in.-thick, 41-ft, 4-in.-long carbon steel pipes. The 220 standard tubes have outside diameters of 28 in. The standard storage tube assemblies, including standard tube plug covers, are described in Section A2.4.3 and shown on Figures A2-10 through A2-15.

Tube base assemblies (see Figure A2-11) locate and attach the base of the storage tubes to the vault basemat. The tube base assemblies attach to basemat embeds as described in Section A4.3.2. Standard storage tube center-to-center distances are the same as the basemat embed center-to-center distances, which are provided in Section A2.4.2.

A4.3.4.3 Functional Requirements. The storage tube and tube base assembly (which is attached to a basemat embed) are required to provide lateral restraint of the MCO during a seismic event. The seismic event is represented by the free field response spectra found on Figure A4-1 and described in WHC-SD-SNF-DB-004 amplified by the response of the building’s attachment points.

The safety function of the standard tubes, which must be maintained during and after a seismic event, is to provide and maintain structural support and the required geometry for cooling. HNF-PRO-097 provides a consistent approach for implementing requirements for design of SSCs designated safety class in accordance with DOE Order 6430.1A.

A4.3.4.4 System Evaluation. The standard storage tubes and base assemblies, attached to the slab embeds, are designed to provide adequate restraint of the MCOs during all NPHs, including seismic events. The restraint provided by the storage tubes maintains the geometric spacing modeled by criticality analyses and required for cooling.

All aspects of the design, fabrication, and installation of the standard storage tube assemblies and tube base assemblies meet Section III of the Boiler and Pressure Vessel Code (ASME 1995), or ANSI/AISC N690-94, or have been demonstrated to be equivalent to Section III (ASME 1995).
The design of the tube body, the tube bottom plate, and the bellows assembly has been shown to meet the requirements of the *Boiler and Pressure Vessel Code*, Section III (ASME 1995). Structural calculations supporting the design, including seismic analysis, predict stresses within allowables.

The tubes were fabricated by rolling plate, joining the plate edges by a longitudinal weld, and attaching the base plate at the bottom and the bellows at the top by a circumferential weld. In Letter 97-SFD-135, *Request for Independent Review Panel (IRP) Concurreance with Nonconformances in the Application of ASME Section III to the Canister Storage Building (CSB) Tube Assemblies to Achieve NRC Nuclear Safety Equivalency* (Hiegel 1997), the tube material and the welds were judged to be equivalent to the *Boiler and Pressure Vessel Code*, Section III (ASME 1995), requirements. The longitudinal weld was inspected by ultrasonic examination in accordance with recommendations resulting from the analysis supporting equivalency. The storage tube bottom plate attachment weld to the storage tube pipe was performed to ANSI/AISC N690-94.

The fabrication requirements of Section III of *Boiler and Pressure Vessel Code* (ASME 1995) were applied to the tube base assemblies (ASME A36 carbon steel). The tube base assemblies provide lateral restraint, which is a structural function. They are bolted to the basemat embeds using ASME SA-307 studs and ASME SA-563 nuts. The structural and thermal functional requirements of Section III, Division 1, Subsection NE, Class MC, of the *Boiler and Pressure Vessel Code* (ASME 1995) have been applied to the standard storage tubes. The exceptions to the application of the *Boiler and Pressure Vessel Code* (ASME 1995) were identified and dispositioned by Letter 97-SFD-135 (Hiegel 1997). The storage tube bellows are fabricated to Section III, Division 1, Article NE-4800 (ASME 1995), for high reliability.

According to WHC-SD-WM-ER-525, *Thermal Hydraulic Feasibility Assessment for the Spent Nuclear Fuel Project*, the vault was designed to ensure that the fuel centerline temperature never exceeds 400 °F under steady-state conditions. This value is substantially less than the 1,112 °F design limit (HNF-SD-SNF-SARR-005). The highest calculated storage tube temperature is 186 °F (Figure A4-3), coincident with a site ambient temperature of 115 °F (Figure A4-2) and a flow rate through the vault of 31,700 ft³/min. These values are from information contained in the addendum to Calculation CSB-HV-0001. The cooling passages in the tube base assemblies provide thermal isolation of the concrete vault basemat and of the steel embeds from the storage tube to ensure that the concrete's temperature limit of 150 °F is not exceeded. In Calculation CSB-SH-1001, *SNF CSB Shielding Source Terms*, radiation levels in the vault are estimated at 200 rem/h for a total integrated dose to the impact absorbers of $7 \times 10^7$ rad. These maximum temperatures and radiation levels are not detrimental to the materials of construction for the standard storage tube assemblies.

The storage tube assembly was investigated for compliance with requirements for operational conditions only without seismic forces. These calculations are included as a reference in Calculation CSB-S-0007A, Appendix K. Modifications to the storage tube assembly design as identified in this investigation were incorporated and form the basis for the design used in this...
current analysis. This analysis supplements the previous one by incorporating the effects of seismic forces.

Operational conditions of the storage tube assembly present during the seismic event included an elevated operating temperature of 220 °F, an operating pressure of 100 lb/in² gauge, plug weight, and gravity. This maximum combined loading condition resulted in a maximum combined stress in the storage tube assembly of 13,400 lb/in². The governing condition included seismic + thermal + gravity forces (no internal pressure). This resulted in a maximum compression stress for design of approximately 12,000 lb/in² (Calculation CSB-S-0007A, Appendix K). The allowable for compression is 17,000 lb/in² according to the requirements of the Boiler and Pressure Vessel Code (ASME 1995), Section III. As addressed in Letter 97-SFD-135 (Hiegel 1997), the maximum combined stress is less than the allowable stress, therefore the standard storage tube assembly design is in compliance with the requirements of the Boiler and Pressure Vessel Code, Section III (ASME 1995).

A4.3.4.5 Controls (Technical Safety Requirements). The standard storage tube and the tube base assemblies are design features, and their safety function is managed by the configuration control process (see Section A5.6).

A4.3.5 Overpack Storage Tubes and Tube Base Assemblies

A4.3.5.1 Safety Function. The overpack storage tubes and tube base assemblies provide the safety functions of preventing damage to MCOs caused by structural failure and providing for natural convective cooling. Confinement of suspect MCOs is a recovery action and is designated safety significant. The overpack storage tubes and tube base assemblies are designated safety class in accordance with DOE Order 6430.1A and ITS Category A using the graded approach of HNF-SD-SNF-DB-003 per the requirements of Sections A3.4.2.1 and A3.4.4.1. The overpack storage tubes and the tube base assemblies provide structural support for the MCOs during NPHs to maintain the geometric array for cooling. A nominal 1-in. clearance between the lower bellows flange and the lower deck embed step (see Figure A2-8) helps to maintain overpack storage tube integrity during a potential MCO eccentric drop accident and to limit the impact damage to the MCO from the drop to ensure that the dropped MCO is retrievable. Safety features of the storage tube design limit potential damage to an MCO and achieve “readily retrievable” for an MCO before and after an accidental drop. This meets the requirements for ready retrievability as defined by NRC equivalency Item 2 of HNF-SD-SNF-DB-003. The overpack storage tubes provide the additional function of containing inert gas and of forming the secondary confinement for the SNF-containing accident-damaged or suspect MCOs. This function is considered a corrective action to an abnormal event.
A4.3.5.2 System Description. The six overpack storage tubes are fabricated from carbon steel rolled plate conforming to ASME SA576 (ASTM A576) or pipe conforming to ASME SA671 (ASTM A671) Gr. CC-70 (ASME 1995, Section II) and are 41 ft, 4 in. long. The overpack storage tubes have an outside diameter of 28 in. Overpack storage tube base assemblies locate and attach the base of the overpack storage tubes to the vault basemat. Overpack storage tube center-to-center distances are the same as the basemat embed center-to-center distances, which are provided in Section A2.4.2. Overpack storage tube base assemblies are constructed of carbon steel plate sections conforming to ASTM A36/A36M, Standard Specification for Carbon Structural Steel. Bottom closure plates on the overpack storage tubes conforming to ASME SA516 (ASTM A516), Grade 70 (ASME 1995, Section II), are 30 in. in diameter to fit snugly into the overpack tube base assemblies. Bellows assemblies consist of a stainless steel, convoluted, thin gauge sheet plate formed into a cylinder conforming to ASME SA-240 (ASTM A240) Type 304 or 321 with stub ends conforming to ASME SA36 (ASTM A36/A36M) or SA516 (ASTM A516/A516M) welded to upper and lower carbon steel flanges conforming to ASME SA516 (ASTM A516/A516M). The lower bellows flange is provided with hold-down bolts to restrain the plug from lifting off the flange surface and releasing radioactive gas or particulate. A separate hold down bolt locking assembly ensures that the plug will not lift. Redundant O-ring seals in the tube plug contain the inert gas and confine the radioactive gas and particulate from the suspect MCO (see Section A2.4.3 and Figures A2-10 through A2-15). Overpack tube plugs are described in Section A4.4.3.

A4.3.5.3 Functional Requirements. The overpack storage tubes and tube base assemblies, which are attached to the basemat embeds, are required to provide lateral restraint of the MCOs during NPHs including seismic events. The seismic event is the free-field response spectra found on Figure A4-1 and described in WHC-SD-SNF-DB-004 amplified by the response of the building's attachment points. The overpack storage tube assemblies are designed to contain the inert gas atmosphere and confine the radioactive gas or particulate that may escape from an off-normal MCO placed in the tube. The design pressure for the overpack tube is 75 lb/in² gauge at a temperature of 220 °F. The pressure vessel safety-class code used for the analysis, design, and fabrication of the overpack storage tube components is the Boiler and Pressure Vessel Code, Section III, Class 2 (ASME 1995). HNF-SD-SNF-DB-003, Item 10, requires that systems and components designated to comply with Section III of the Boiler and Pressure Vessel Code implement NRC Regulatory Guide 1.26 in assigning the appropriate code class. Subsection NC has been selected for the overpack storage tube assemblies because Subsection NC provides more stringent quality requirements and is generally applied to containment vessels. Similarly, the other components of the overpack storage tube assemblies also are assigned Subsection NC based on using Category B from NRC Regulatory Guide 1.26. Quality Category B is based on the overpack storage tube providing postaccident containment and heat removal in accordance with NRC Regulatory Guide 1.26. The structural safety-class code used for the analysis, design, and fabrication of the overpack storage tube base assembly is ANSI/AISC N690-94. For continuity of design and future operational flexibility, the safety-significant overpack tube plugs are designed to the Boiler and Pressure Vessel Code, Section III, Class 2, Subsection NC requirements (ASME 1995), as are the safety-class overpack storage tube assemblies.
A4.3.5.4 System Evaluation. The overpack storage tubes and tube base assemblies, attached to the basemat embeds, are required to provide adequate restraint of the MCOs during all NPHs, including seismic events. The overpack storage tubes are designed to the *Boiler and Pressure Vessel Code*, Section III, Division 1, Subsection NC (ASME 1995). CSB-S-0007, *Storage Tube Analysis and MCO Drop*, documents a structural analysis of the overpack storage tube base assembly. The analysis results show that the overpack tube base assembly design provides adequate support for the overpack tube assembly and MCO during accident drop conditions and postulated DBE and NPH events.

The interaction of the MCO with the standard storage tube during a DBE has been analyzed to determine the maximum forces generated by the MCO storage tube on the vault's operating deck and basemat and to determine the maximum stresses developed in the storage tube (see Section A4.3.4). Because the design of the overpack storage tube assemblies is similar to the standard storage tube assemblies, the design of the overpack storage tube assembly is in compliance with the seismic requirements of the *Boiler and Pressure Vessel Code*, Section VIII (ASME 1995). *Boiler and Pressure Vessel Code*, Section VIII, allows are lower than the allowables in Section III, Class 2, Subsection NC. Therefore the overpack storage tube assemblies exceed the seismic requirements of the *Boiler and Pressure Vessel Code*, Section III (ASME 1995).

Overpack storage tube assemblies are required to be pneumatically pressure tested by the fabricator in accordance with the requirements of *Boiler and Pressure Vessel Code*, Section III, Division 1, Subsection NC, Article NC-6300 (ASME 1995), using air or inert gas at a pressure of 94 lb/in² gauge.

The vault is designed to ensure that the fuel centerline temperature never exceeds 400 °F under steady-state conditions (WHC-SD-WM-ER-525). This value is substantially less than the 1,112 °F design limit (HNF-SD-SNF-SARR-005). The highest calculated storage tube temperature is 186 °F (Figure A4-3), coincident with a site ambient temperature of 115 °F (Figure A4-2) and a flow rate through the vault of 31,700 ft³/min. These values are from information contained in the addendum to Calculation CSB-HV-0001 and are conservative given that only one MCO is in an overpack storage tube assembly. The cooling passages in the tube base assemblies provide thermal isolation of the concrete vault basemat and of the steel basemat embeds from the storage tube to ensure that the concrete's temperature limit of 150 °F is not exceeded.

A4.3.5.5 Controls (Technical Safety Requirements). The overpack storage tube and the tube base assemblies are design features, and their safety function is managed by the configuration control process (see Section A5.6). There are no TSR controls for this SSC because its safety function is related to recovery actions, not to accidents during normal operations. Controls related to specific recovery actions must be developed on a case-by-case basis depending on current conditions and recovery evaluation results.
A4.3.6 Canister Storage Building Intake Structure and Exhaust Stack

A4.3.6.1 Safety Function. The safety function of the CSB intake and exhaust stack is to provide passive cooling for MCOs. The intake structure and exhaust stack are classified as safety class in accordance with DOE Order 6430.1A and ITS Category A using the graded approach of HNF-SD-SNF-DB-003 per the requirements of Section A3.4.2.6. They are integral parts of the CSB passive cooling system and, as such, are designed for the DBE and design basis tornado wind loads.

A4.3.6.2 System Description. The CSB intake structure is a 17 ft by 18 ft by 22 ft high welded steel structure weighing 867,000 lb that is mounted on top of a 15 ft by 20 ft by 57 ft high reinforced concrete tower constructed monolithically with the CSB vault's intake plenum. The steel portion of the intake structure (tower) is constructed as a braced tubular steel frame with steel shear panels at the base. Tubular steel members conform to ASTM-A588/A588M, Standard Specification for High-Strength Low-Alloy Structural Steel with 150 Ksi (345 MP), Grade A with a minimum yield strength of 50,000 lb/in². Space cloth Type 304 stainless steel covers the intake structure and acts as a debris barrier for the CSB vault's convective cooling system. Reinforced concrete is specified to have a 28-day compressive strength of 4,000 lb/in² in accordance with ANSI/ACI 349-85. Reinforcing steel is in accordance with ASTM A615/A615M-96a Grade 60.

The exhaust stack is a self-supporting, 169-ft-tall, cantilevered structure. The diameter at the base of the stack is 13 ft, and the diameter at the top of the stack is 7 ft. The bottom section of the 13-ft-diameter stack base rises 67 ft above floor level and is constructed of 0.63-in.-thick steel plate. The bottom section of the exhaust stack is attached to a reinforced concrete foundation 20 ft, 9 in. in diameter and 12 ft, 1.5 in. thick. The middle section of the exhaust stack continues upward 33 ft and tapers from 13 ft to 7 ft in diameter. The middle section is constructed of 0.63-in.-thick steel plate. The 7-ft-diameter top section of the exhaust stack continues upward another 57 ft and is constructed of 0.38-in.-thick steel plate. The upper 44 ft of the top section of the stack are provided with a series of vertical wind spoilers that protrude 7 in. from the stack and are each 11 ft long. The spoilers are made of 0.38-in.-thick steel plate and are designed to the guidance in ASCE-7-93. Stack material meets the specifications in ASTM A847, Standard Specification for Cold-Formed Welded and Seamless High Strength, Low Alloy Structural Tubing with Improved Atmospheric Corrosion Resistance, ASTM A606, Standard Specification for Steel, Sheet and Strip, High-Strength, Low-Alloy, Hot-Rolled and Cold-Rolled, with Improved Atmospheric Corrosion Resistance, or ASTM A588/A588M Grade A (50,000 lb/in² yield). The stack is topped by a 12-ft-tall wind deflector as shown on Figure A2-1.

A4.3.6.3 Functional Requirements. The intake structure and exhaust stack for the CSB vault must provide airflow adequate to prevent overheating of the SNF. This overheating could be caused by damage to the structure from the DBE, tornado winds, or other NPHs such as ashfall. A summary of loads considered in the design of the intake structure and exhaust stack is given below.
Annex A — Canister Storage Building

- **Intake structure**
  - Roof and structure dead load
  - Roof live load (20 lb/ft²)
  - Snow load at ground level
  - Extreme wind (90 mi/h)
  - Tornado wind (200 mi/h)
  - DBE seismic free-field response spectra applied at base of supporting tower analyzed as part of the vault-CSB model
  - Volcanic ashfall ground load

- **Exhaust stack**
  - Structure dead load
  - Extreme wind (100 mi/h)
  - Tornado wind (200 mi/h)
  - DBE free-field response spectra applied at base of stack.

The structural safety-class code used for the analysis, design, and fabrication of the intake structure and exhaust stack components is ANSI/AISC N690-94 for the structural steel intake and exhaust stack portions, and ANSI/ACI 349-85 for the reinforced concrete intake towers.

**A4.3.6.4 System Evaluation.** The intake structure and exhaust stack are designed to survive the DBE and tornado wind loads given in Table A1-1.

The intake structure (tower) has been designed and analyzed as an integral part of the CSB vault structure (Calculation CSB-S-0028A). In the vault gravity and seismic analysis, the intake housings were modeled as lumped masses located at the proper elevation on top of the concrete towers, which were modeled using plate elements. The reinforced concrete towers were designed based on the loads from the vault gravity and seismic load models. In-structure response spectra at the base of the towers enveloping one-half mean, mean, and two times mean soil properties were developed using the vault seismic model. To design the steel housing, a single model was created that included a detailed model of the steel housing on top of a stick model of the intake tower. The enveloped response spectra at the base of the towers developed from the vault model
Post-construction measurements and analysis of the intake structure and operating structure are documented in Calculation CSB-S-0048, Supplement H, *Operating Shelter Roof Dead Load Confirmatory*. The analysis confirms that it is possible for the CSB operating shelter structure to impact the intake structure during a DBE. The estimated intake structure top deflection added to the estimated operating structure deflection at the same height is 5 in., and the as-built gap between the siding and the intake structure varies from 4.75 in. to 5.375 in. Analysis of the impact of the roof deck with the concrete intake structure has determined that no permanent effect on the intake structure results. The roof deck is estimated to buckle locally where it impacts the intake structure, and the damage is estimated to be slight and should not compromise the safety-significant function of the operating area structure (see Section A4.4). The estimated deflections in Table A4-4 are derived using conservative assumptions. Use of actual values in the estimates would probably preclude impact during a seismic event. Actual values were not used in the estimates because the analysis showed that the slight buckling effect in the conservative analysis does not compromise the safety-significant function of the operating area structure.

Table A4-4. Deflections at Top of Intake Structure and Exhaust Stack.

<table>
<thead>
<tr>
<th></th>
<th>Tornado loading</th>
<th></th>
<th>Seismic loading</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>East-west (in.)</td>
<td>North-south (in.)</td>
<td>Vertical (in.)</td>
</tr>
<tr>
<td>Intake structure</td>
<td>0.04</td>
<td>0.02</td>
<td>0.00</td>
</tr>
<tr>
<td>top of concrete</td>
<td></td>
<td></td>
<td>Enveloped</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>response spectra</td>
</tr>
<tr>
<td>Intake structure</td>
<td>0.10</td>
<td>0.07</td>
<td>0.01</td>
</tr>
<tr>
<td>top of steel</td>
<td></td>
<td></td>
<td>0.25</td>
</tr>
<tr>
<td>Exhaust stack</td>
<td>13.37</td>
<td>13.31</td>
<td>0.03</td>
</tr>
<tr>
<td>top of steel stack</td>
<td></td>
<td></td>
<td>1/2 x mean</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>4.95</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>4.96</td>
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<td>0.03</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>7.00</td>
</tr>
<tr>
<td></td>
<td>Mean</td>
<td>5.43</td>
<td>5.35</td>
</tr>
<tr>
<td></td>
<td>2 x mean</td>
<td>5.26</td>
<td>5.34</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.02</td>
<td>7.50</td>
</tr>
</tbody>
</table>

SRSS = square root of the sum of the squares.

The exhaust stack was analyzed for tornado loading in Calculation CSB-S-0012. The internal loads and moments for the 200 mi/h tornado load were developed, and the stack design was modified as required. A confirmatory analysis of the exhaust stack was performed in Calculation CSB-S-0045, *Exhaust Stack Pedestal Analysis*. The general procedure was to compare shears and moments for the exhaust stack that were obtained from the three-dimensional SAP90 vault model (CSI 1995) with the design shears and moments for tornado from Calculation.
CSB-S-0012. The exhaust stack results for the one-half mean, mean, and two times mean soil property analyses of the vault were compared with the tornado design loads from Calculation CSB-S-0012. It was found that the tornado loads were 55% higher at the base of the stack and 14% higher at the bottom of the 7-ft-diameter section. Based on these results, the stack was deemed adequate for the revised seismic loads. Thermal gradients across the stack are of minimal design concern for the materials and temperatures involved.

To design the steel housing, a single model was created that included a detailed model of the steel housing on top of a stick model of the intake tower. The estimated response spectra at the base of the towers developed from the vault model were used as input motion for the intake structure-tower model. Gravity and tornado loads were applied as static loads in the design of the steel structure. The estimated deflections representative of tornado wind and seismic loads are listed in Table A4-4.

A heat transfer evaluation of the air intake structure, the underground vault airflow paths, and the exhaust stack concluded that the CSB natural convection cooling design provides adequate cooling of the MCOs and maintains the MCO surface temperatures and concrete temperatures within the allowable limits (Calculation CSB-HV-0001). This heat transfer analysis is based on vault 1 having a total heat load of 161.2 kW. This is the heat load associated with the current inventory of SNF in the K Basins. The heat transfer calculations used three computer codes for the analysis.

- The T-Duct\(^1\) simulation program allowed the analyst to model the airflow system (inlet screen, inlet stack, ducts, duct changes in size, the vault inlet plenum, the staggered tube bank in the vault, the vault exhaust plenum, and the stack size and height) and provide airflow rates and system pressures using the two basic principles, conservation of mass and conservation of energy.

- The PHOENICS\(^2\) program analyzed the temperature and velocity profiles within the storage vault by solving equations for flows, turbulence, mass, and heat transfer using a controlled volume method based on the finite element approach.

- A LOTUS spreadsheet was used to model the “hot spot” in the vault, the area of the vault where, because of the buoyancy effect and air turbulence, the air temperatures are the highest. This spreadsheet solved a series of nonlinear simultaneous algebraic heat transfer equations using the matrix inversion method. The computer code analyzed the surface temperature of the MCO and the resultant maximum concrete temperature.

\(^{1}\)T-Method Duct Design computer code, Fluor Daniel, Incorporated, Irvine, California.
\(^{2}\)PHOENICS computer code, CHAM of North America, Huntsville, Alabama.
Considered in these analyses were important factors that influence the design of the cooling system in limiting the MCO surface temperature (e.g., ashfall, hoarfrost, wind effect, and loading scenario). Loss of natural convective ventilation as a result of “intake blockage” is not considered credible because the building has a safety-class elevated air intake structure. The following statements summarize the heat transfer analysis.

- The air intake structure using the cruciform and canopy design creates a positive pressure at the intake under all wind directions. In addition, the deflector at the top of the exhaust stack enhances exhaust airflow by increasing the wind suction pressure for all wind directions.

- The MCO maximum external temperature will be maintained below 270 °F as long as the total decay heat generated by the stored SNF is limited to 161.2 kW and the maximum heat released by an MCO is 800 W.

- The elevated 7-ft-diameter vault exhaust stack at 169 ft above the operating deck provides adequate buoyancy forces (the stack effect) for inducing airflow.

- The sequence of MCO loading into the vault tubes is optional, however the preferred loading is that the storage tubes near the exhaust stack be loaded first.

- When vault 1 is fully loaded, the long-term concrete temperature has been calculated to remain well below the allowable maximum concrete temperature (150 °F) required by ANSI/ACI 349-85, Appendix A, for long-term normal operation.

CHAM of North America, the author of the PHOENICS computer program, stated in *PHOENICS Computer Program* (CHAM 1991), that the code “is very well validated and developed from over 10 years of commercial use.” Monitoring and verification of the results of the CSB heat transfer and flow calculations using PHOENICS were performed using the Design Control Procedures Manual (Section 4, “Calculations”, and Section 9, “Computer Code Validation and Verification”). The following steps were included: (1) available commercial computer programs were studied and the most efficient, PHOENICS, was selected; (2) the PHOENICS introductory course, given by CHAM of North America, was completed by the heat transfer analyst; (3) the Computer Code Summary sheet has been completed and verified; (4) sample test problems were run by the heat transfer analyst before using the program; (5) a study of the publications where the PHOENICS program was verified by tests was conducted; (6) a complete description of the model was included in the report (CHAM 1991); and (7) an independent reviewer checked and verified that the model used is a reasonable representation of the CSB physical constraints and the outputs are reasonable.

Although a three-dimensional vault thermal analysis was planned, the PHOENICS 2D Computational Fluid Dynamics model has been used showing that the temperature difference between highest value (hot spot) and the outlet air temperature is only 4 °F (CSB-HV-0001, page 41, Figure 6A). The Los Alamos National Laboratory obtained a similar conclusion for their
vault as a result of three-dimensional modeling using the CFX computer code. Their conclusion is documented in "Conceptual Design Report for Line Item Project Nuclear Materials Storage Facility Renovation at the Los Alamos National Laboratory, Technical Area 55, Building PF-41" (LANL 1995).

Validation of the T-Duct computer calculations was performed by comparing the results of manual calculations with computer output for duct systems including five-section and nineteen-section systems. The five-section comparison provided verification of the airflows, air velocities, friction factors, pressure losses, and mass flow rates balancing. The nineteen-section comparison verified C-coefficients selection. The comparisons revealed insignificant differences, demonstrating the validity of the T-Duct code.

The computer calculations using LOTUS were verified by comparison with hand calculations to determine the maximum MCO and concrete temperatures in the vault hot spot. All equations used in the matrix were verified.

The vault cooling system is an entirely passive system. No mechanical or electrical equipment is required for it to perform its function. There are no identifiable hazards related to operation of the vault cooling system. Airflow through the vault is enhanced by the structure of the exhaust plenum because the top of the exhaust plenum is 7 ft higher than the top of the inlet plenum.

BNFL Inc. performed an independent review of the natural convection cooling calculations provided by Fluor Daniel, Incorporated. The BNFL Inc. report (see CSB-HV-0001, Appendix H) agreed with the basic conclusions of the Fluor Daniel, Incorporated, heat transfer analysis and also offered some constructive comments on issues associated with the design of the CSB as follows.

- The extreme ambient temperature of 115 °F used in the analysis has been recorded at the Hanford Site once in the past 50 years for a 2-hour duration. Modeling the CSB using this high temperature provides a safety margin of about 3 °F to 5 °F for the maximum temperatures calculated.

- The one-dimensional heat balance analysis used to calculate the maximum temperature (MCO, tube, and concrete) uses a heat balance of a tube located in the hottest part of the vault and loaded with two MCOs at 800 W each. The calculation shows the maximum concrete temperature (121 °F) is substantially lower than the 150 °F concrete temperature limit.

- The heat transfer analysis used a two-dimensional model (PHOENICS computer code) of the vault from which the predominant hot spots were determined. It is expected that the results of a three-dimensional model (which would include eddy current airflow patterns within the vault) would show lower temperatures, providing additional margins of conservatism.
The CSB model used in the heat transfer analysis assumed that the vault walls were smooth with no obstructions impeding airflow around the storage tubes. This flow pattern increases the tube temperature because some of the air would bypass the tubes and not provide direct cooling of the tubes. As Figure A2-5 indicates, 10 air foil baffles have been installed on the walls of vault 1 to impede airflow that would bypass the storage tubes. These baffles direct air in a cross flow pattern around the storage tubes and thus increase heat transfer from the storage tubes to the air. This provides an additional measure of conservatism to the maximum temperatures calculated for the heat transfer analysis.

Wind tunnel tests at Colorado State University, Fluid Dynamics and Diffusion Laboratory, demonstrated that wind flow direction can affect the performance of the CSB passive cooling system. The laboratory simulated airflow patterns around the CSB using a scale model of the CSB, including surrounding structures, in their laboratory wind tunnel. The results of the tests showed that for all wind directions, a net positive pressure occurs at the capped cruciform/screen inlet structure of the CSB inlet stack. The results also showed that for all wind directions, a net negative pressure occurs at the deflector mounted on top of the CSB exhaust stack. Thus, for the CSB intake cruciform and exhaust stack deflector designs, wind from any direction enhances the pressure gradient caused by the passive natural convective cooling system, increases the airflow rate through vault 1, and provides further conservatism to the maximum temperatures calculated for the heat transfer analysis.

A4.3.6.5 Controls (Technical Safety Requirements). The CSB above-grade structures (intake structure and exhaust stack) are design features (see Section A5.6), and their safety function is managed by the configuration control process.

A4.3.7 Multi-Canister Overpack

A4.3.7.1 through A4.3.7.4 The MCO has been classified as safety class in accordance with DOE Order 6430.1A and ITS Category A using the graded approach of HNF-SD-SNF-DB-003. The safety functions for the MCO are providing confinement of SNF and maintaining geometrical arrangements assumed in criticality analyses. See the MCO Topical Report (HNF-SD-SNF-SARR-005) for the safety functions, system description, functional requirements, and system evaluation of the MCO.

The MCO is a single-use container consisting of the following components:

- Stainless steel shell
- Five or six fuel baskets containing SNF
- None, one, or two scrap baskets containing SNF
- Stainless steel center post (Mark IA baskets only)
- Stainless steel shield plug
- Stainless steel locking ring.
These components and configurations are shown on Figure A2-2 and described in the MCO Topical Report (HNF-SD-SNF-SARR-005).

The interface between the MCO and the MHM grapple is at the MCO ring collar and canister cover assembly. As described in the MCO Topical Report (HNF-SD-SNF-SARR-005), the design of the MCO ring collar uses the guidance of ANSI N14.6-1993 of a factor of safety of three on material yield and five on material ultimate strength. The canister cover assembly design uses the normal (Service Level A) criteria of Subsection NB of Section III of the Boiler and Pressure Vessel Code (ASME 1995). A mechanically sealed MCO may have a leakage rate of no greater than $10^{-5}$ cm$^3$/s as defined by the MCO Topical Report (HNF-SD-SNF-SARR-005).

The canister cover assembly must support the total weight of the MCO and contents for lifting, which equates to a conservative total lifting capacity of 12 tons. A lifting grapple with six gripping shoes will be used to lift the MCO and its contents by the canister cover assembly. Figure A2-38 displays the gripping shoe configuration for the lifting grapple-canister cover assembly interface. The stress through the MCO shell resulting from the lifting load is 615 lb/in$^2$. The thinnest point in the shell is located at the thread relief in the canister collar. Since it will also see the lifting load through its section, the thread relief was analyzed and found to have a stress of 866 lb/in$^2$.

The canister cover assembly must be able to withstand an internal pressure of 450 lb/in$^2$ gauge. The MCO shell has a thickness of 0.5 in., and its inside diameter is 23 in. The stress through the shell due to the pressure load was found to be 10,575 lb/in$^2$. The stress for the thinnest section through the canister collar buttress thread (0.354 in. thick) is 15,840 lb/in$^2$.

When both lifting and pressure loads are applied together, the stress through the thinned portion of the canister cover assembly shell due to these loads then becomes:

- Axial stress = 39,200 lb/in$^2$
- Axial stress ratio = 0.88
- Hoop stress = 8,300 lb/in$^2$
- Hoop stress ratio = 0.19.

A4.3.7.5 Controls (Technical Safety Requirements). The MCO is a design feature (see Section A5.6) and its safety function is managed by the configuration control process.

A4.3.8 Transportation Cask

A4.3.8.1 Safety Function. The transportation cask is a safety-class equipment item as defined at other facilities and provides a safety-significant function for the CSB. The transportation cask has the safety function of providing a confinement barrier and structural protection to the MCO from drops from the receiving crane.
A4.3.8.2 System Description. The transportation cask consists of a body fabricated from stainless steel forgings conforming to ASME SA-336 (ASTM A336/A336M) Type F304, and a bolted-on stainless steel lid conforming to ASME SA-336 Type F304 with two welded-on trunnions conforming to ASME SA-336 Type F304 or SA-240 Type 304. The overall dimensions of the cask are 170.5 in. long and 39.81 in. in diameter. The cask cavity is 160.5 in. long and has an inner diameter of 25.19 in. Other dimensions and details of the transportation cask can be found in HNF-SD-TP-SARP-017, Safety Analysis Report for Packaging (Onsite) Multi-Canister Overpack. The SNF FSAR CSB Annex assumes the design and configuration shown in Revision 1 of HNF-SD-TP-SARP-017. The transportation cask is designed to be transported in the vertical position by a dedicated semitrailer. This arrangement is shown on Figure A2-20.

A4.3.8.3 Functional Requirements. The transportation cask is required to provide a confinement barrier and structural protection from drops from the receiving crane.

A4.3.8.4 System Evaluation. The transportation cask has been analyzed for its capability to provide the necessary protection of the MCO during cask lowering by the receiving crane and a collision of the MHM with the receiving crane. The analysis conservatively eliminated any energy absorption contribution by the receiving crane or MHM. In the analysis the MHM was modeled as a rigid body. Upon contact with the cask, the MHM was further driven into the cask at the maximum drive wheel torque, resulting in additional loading on the cask wall of 6,000 lb. The cask was conservatively positioned, impacting the rigid cask receiving pit edge at its midpoint. The transportation cask, MCO, Mark IA baskets, lifting fixture, simplified MHM, and rigid transfer pit were modeled dynamically by finite element analysis.

Analysis of the impact at the cask receiving pit was broken up into two dynamic simulations: the first simulated the MHM impacting the static cask–MCO suspended by the lifting fixture, and the second simulated the MHM and cask–MCO moving sideways at 40 ft/min and impacting the edge of the cask receiving pit. The results of the first simulation indicate that the cask, MCO, and all six Mark IA baskets remain linearly elastic; there is no plastic damage. The results of the MHM and cask–MCO impacting the cask receiving pit edge indicate that the MHM is stopped and rebounds. The maximum equivalent plastic strain was in the bottom basket with a value of approximately 0.3% strain. There was no significant plastic damage. These results are contained in analyses documented in transmittal FDT-137, MHM Collision with Cask/MCO (Peterson 1998).

A4.3.8.5 Controls (Technical Safety Requirements). The transportation cask is a design feature and requires no TSR controls to ensure performance of its safety function (see Section A5.6.1).
A4.3.9 Multi-Canister Overpack Handling Machine Seismic Restraint System

A4.3.9.1 Safety Function. The safety function of the MHM seismic restraint system is to prevent shearing of an MCO by the MHM during a DBE and during all MCO handling activities so that the MHM cannot apply significant shearing stress to an MCO that is partially lowered from the MHM’s MCO cavity. The seismic restraint system has a related safety function of ensuring that the MHM will not impact or structurally collapse and cause damage to the safety-class operating deck. The MHM seismic restraint system is designated safety class in accordance with DOE Order 6430.1A and ITS Category A in accordance with the graded approach in HNF-SD-SNF-DB-003.

A4.3.9.2 System Description. The MHM seismic restraint system consists of four subsystems: the bridge seismic restraint system, the trolley seismic restraint system, the turret locking pin system, and the base locking pin system. These systems restrain movement of the MHM turret with respect to the trolley and nose assembly and translational movement of the MHM trolley and bridge with respect to the facility.

The MHM bridge seismic restraint system comprises spring-applied rail clamps, bridge wheels, shaped hooks that are located under the railhead, and a motor and hydraulic components that release the rail clamps. The bridge seismic clamps physically interface with the MHM bridge rails installed on the CSB operating deck. The rail clamps have fail-safe features in which the shoes are applied by coil springs and the hydraulic motors are energized to disengage the clamps. The rail clamps provide Y (north-south) direction seismic restraint. The wheel flanges provide X (east-west) direction seismic restraint. The Z (vertical) direction seismic restraint is provided by the shaped hooks positioned under the railhead to prevent vertical uplift.

The trolley seismic restraint system is composed of X-restraint pins, pin actuation motors, screw jacks and jack drive, steel pockets on the bridge beams, trolley wheels, and lugs that hook beneath the bridge beam top flange. Two X-restraint pins lock the trolley to the bridge rails, preventing east–west motion of the trolley and turret. The trolley pins are inserted by the pin actuation motors into pockets welded onto the top of each bridge beam. The screw jacks and jack drive allow the exact positioning of the pins to be adjusted so that the pins line up with the corresponding bridge beam pockets. Restraints for the trolley in the Y direction are passive restraints formed by the trolley wheel flanges that mate to the trolley rails. Passive restraints for the Z direction are provided by four lugs that hook beneath the top flange of each beam.

The turret locking pin system consists of the turret locking pin, an actuation screw jack and motor, and three steel pockets installed on the turret support turntable. The actuation screw jack and motor are installed on the MHM trolley. The turret locking pin is a 5.1-in.-diameter steel pin that fits into any of three pockets on the turret support turntable. When the pin is engaged in one of these three pockets, it serves to precisely align the turret with one of its three positions (tube plug, MCO, or camera) and to prevent the turret from moving by securing it to the trolley.

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A powered, base locking pin system consists of the base locking pin, an actuation screw jack and motor, and three steel pockets in the base nose block. This system is similar to the turret locking pin system except that it secures the turret base plate to the machine base nose block at each turret index position. A base locking pin locks the turntable to the lower turret.

Visual verification of the four seismic restraint subsystems ensures that they are properly positioned and capable of performing their designed seismic restraint functions. Verification that the seismic restraints are in position is procedurally required and will be surveyed before MCO hoist operation.

A4.3.9.3 Functional Requirements. The MHM seismic restraint system, when applied, is required to limit the translational movement of the bridge, trolley, and turret that could cause significant physical damage to an MCO within or partially within the MHM. All of the components that provide restraint in the Z direction are always engaged (trolley uplift hooks, bridge hooks). The bridge X direction (wheel flanges) and the trolley Y direction restraints are passive and always effective. If not engaged, the bridge seismic rail clamps slowly actuate after a loss of power. All of the seismic structural components are required to safely withstand seismic forces during a DBE and prevent the MHM from collapsing onto the operating deck. The seismic input is based on the response spectra included in Figure A4-4.

A4.3.9.4 System Evaluation. All of the seismic structural components have the strength to safely withstand seismic forces during a DBE. The MHM bridge seismic clamps (four total) are each capable of applying a 200,000 lb holding force to the 175 lb per yard crane rails. Loading conditions used to generate the seismic analyses were evaluated in accordance with the requirements of ASME NOG-1-1995. Analyses documented in ESL/R(96)083, Hanford MHM Seismic Analysis of the Hanford MCO Handling Machine, demonstrate that the MCO will not be trapped while being lowered or retrieved by the MHM during the DBE. The seismic analysis report [ESL/R(96)083] provides the results of seismic analysis for the MHM bridge wheel and trolley wheel seismic restraints (Table A4-5).

A4.3.9.5 Controls (Technical Safety Requirements). The MHM seismic restraint system is a design feature of the MHM (see Section A5.6), and its safety function is managed by the configuration control process. The MHM vendor has not established any special maintenance or surveillance requirements. The following TSR is required to ensure that the seismic restraint system will perform its safety function.

- Visual verification that the east and west bridge clamps engage the rails (hydraulic pressure gauges indicate a low pressure), the seismic locking pins for the North and South trolley rails are inserted, the turret locking pin is inserted, and the base locking pin is inserted must be performed before MCO hoist operation (see Chapter A5.0, Section A5.5.3.3).
Table A4-5. Maximum Multi-Canister Overpack Handling Machine Trolley Wheel and Bridge Wheel Seismic Restraint Loads.

<table>
<thead>
<tr>
<th>Restraint load</th>
<th>Final seismic analysis (kilonewtons)</th>
<th>Ederer design basis (kilonewtons)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Dynamic across rail at a single wheel</td>
<td>566.4</td>
<td>568.0</td>
</tr>
<tr>
<td>Combined (downward) at a single wheel</td>
<td>1480.8</td>
<td>1495.7</td>
</tr>
<tr>
<td>Combined (upward) at a single hook</td>
<td>430.6</td>
<td>551.0</td>
</tr>
<tr>
<td>Total dynamic along one rail</td>
<td>1509.0</td>
<td>1588.0</td>
</tr>
<tr>
<td>Dynamic across rail at a single end truck pivot pin</td>
<td>1305.0</td>
<td>1430.0</td>
</tr>
<tr>
<td>Combined (downward) at a single end truck pivot pin</td>
<td>2468.0</td>
<td>2575.0</td>
</tr>
<tr>
<td>Combined (upward) at a single end truck pivot pin</td>
<td>66.0</td>
<td>440.0</td>
</tr>
</tbody>
</table>

A4.3.10 Multi-Canister Overpack Handling Machine Rails and Rail Frogs

A4.3.10.1 Safety Function. The safety function of the MHM rails and rail frogs is to prevent shearing of an MCO by restraining movement of the MHM. Failure to withstand the seismic loads could lead to inadvertent movement and collapse of the MHM, which could damage CSB facility structures designated safety class. Because they function to ensure operation of the MHM seismic restraints during the DBE, the MHM rails and rail frogs are designated safety class in accordance with DOE Order 6430.1A and ITS Category A in accordance with the graded approach in HNF-SD-SNF-DB-003.

A4.3.10.2 System Description. The MHM rails and rail frogs provide for safe movement of the MHM and facilitate safe transport of the MCOs throughout the CSB facility. Seismic rail clamps lock the MHM wheel trucks to the rails whenever the MHM is not in transport MCO mode, and they are locked automatically by a signal from the seismic detection and MHM power-disconnect actuation system.

The MHM rails are located at the east and west edges of the operating area deck directly over the thick vault walls. The two parallel rails run in a north-to-south direction from the load-in/load-out area in the north to the sampling/weld station in the south. The rails are recessed into trenches to maintain the top of the rail at deck level. The trenches are wide enough to permit clamping of the MHM rail clamps.
The MHM rails transmit their loads to the operating area deck through a system of bearing plates, rail clips, lateral restraints, and anchor bolts. The bearing plates are provided with threaded holes. Anchor bolts installed into the reinforced concrete of the operating area deck secure the bearing plates. Bolted connections through the rail clips secure the rails to the bearing plates. Lateral restraint is provided by keeper bars placed alongside the toe of the rail between rail clips. The keeper bars are welded to the bearing plate.

Rail frogs are a welded fabrication of rails and base plates that bolt directly to the rail bearing plate. The rail frogs provide a means for the MHM and receiving crane to alternately access the cask receiving pit. The rail frogs incorporate design features to activate trip sensors for the MHM and hard stops for both cranes. The rail frogs are aligned, positioned, and secured manually to administratively limit the load-in/load-out area to one crane at a time.

MHM and rail frog rails are constructed of 175 lb/yd rails conforming to American Institute of Steel Construction configuration; the continuous-welded material conforms to ASTM A759-85, Standard Specification for Carbon Steel Crane Rails. Bearing plates are constructed of 1-in.-thick plate material conforming to ASTM A36/A36M. Rail clips are constructed of 1.5-in.-thick material conforming to ASTM A572/A572M, Standard Specification for High-Strength Low-Alloy Columbium-Vanadium Structural Steel, Grade 70. Anchor bolts are 1.25-in.-diameter threaded stud anchors conforming to ASTM A449-93, Standard Specification for Quenched and Tempered Steel Bolts and Studs, with an embedment length of approximately 24 in. Keeper bars are constructed of 0.75-in.-thick, long carbon steel conforming to ASTM A572/A572M Grade 50. Fabrication and manufacturing of the rail frogs are in accordance with AWS D14.1, Specifications for Welding of Industrial and Mill Cranes and Other Material Handling Equipment. Rail runways are flashbutt welded at the job site using procedures developed in accordance with AWS D14.1.

A4.3.10.3 Functional Requirements. Nominal and seismic loads are imposed on the rails and rail frogs. Live, dead weight, and seismic loads caused by the MHM response are transmitted to the reinforced concrete of the operating deck for dissipation into the below-grade vault walls. Acceptable analytical results are required in order to demonstrate that the rails and rail frogs withstand and transmit these load combinations.

DOE Order 6430.1A provides criteria that SSCs designated safety class be designed in accordance with the Boiler and Pressure Vessel Code, Section III, Class 2 (ASME 1995), or other applicable nuclear safety code. HNF-PRO-704 refers to HNF-PRO-097 for design codes and standards to safety-class structural SSCs. HNF-PRO-097 provides a consistent method for implementing these codes and standards. HNF-PRO-704 requires that safety-class structural SSCs be designed and constructed in accordance with ANSI/AISC N690-94. In order to demonstrate MHM rail and rail frog structural availability, independent checks of hold-down bolt torque must be performed each time the rail frogs are repositioned.

A4.3.10.4 System Evaluation. The governing loads were used to calculate the MHM rail and rail frog component loads and to determine material properties, thicknesses, and spacing for the...
rails, anchor bolts, bearing plates, base plates, rail clips, keeper bars, and hold down bolts. 

Governing MHM crane rail loads were obtained from the MHM modeling using ANSYS 
BEAM44 and BEAM44 elements for the trolley at the mid-span, quarter span, and mid position 
following the requirements of ASME NOG-1-1995, Section NOG-4151. The seismic analysis 
report [ESL/R(96)083] provided the results of these analyses. These results are included in 
Table A4-5. The demand/capacity ratios for various MHM components associated with the 
seismic restraints are shown in Table A4-6.

<table>
<thead>
<tr>
<th>Component</th>
<th>Demand/capacity</th>
<th>Supplier</th>
</tr>
</thead>
<tbody>
<tr>
<td>Trolley uplift hooks (welds)</td>
<td>0.98</td>
<td>Ederer</td>
</tr>
<tr>
<td>Bridge uplift hooks</td>
<td>0.72</td>
<td>Ederer</td>
</tr>
<tr>
<td>Johnson clamps</td>
<td>0.85</td>
<td>Ederer</td>
</tr>
<tr>
<td>End tie beam lug</td>
<td>0.89</td>
<td>Ederer</td>
</tr>
<tr>
<td>Trolley rail support beam (weld)</td>
<td>0.99</td>
<td>Ederer</td>
</tr>
<tr>
<td>Girder splice plate</td>
<td>0.97</td>
<td>Ederer</td>
</tr>
<tr>
<td>Girder flange</td>
<td>0.94</td>
<td>Ederer</td>
</tr>
<tr>
<td>Girder splice bolt #5</td>
<td>0.90</td>
<td>Ederer</td>
</tr>
<tr>
<td>Overall trolley stresses</td>
<td>0.91</td>
<td>Ederer</td>
</tr>
<tr>
<td>Overall bridge stresses</td>
<td>0.76</td>
<td>Ederer</td>
</tr>
<tr>
<td>Overall turret stresses</td>
<td>0.59</td>
<td>GEC</td>
</tr>
<tr>
<td>Overall turntable stresses</td>
<td>0.59</td>
<td>GEC</td>
</tr>
<tr>
<td>Overall end carriage stresses</td>
<td>0.83</td>
<td>Ederer</td>
</tr>
<tr>
<td>Base locking pin (bending limiting)</td>
<td>0.2</td>
<td>GEC</td>
</tr>
<tr>
<td>Turret locking pin (bending)</td>
<td>0.27</td>
<td>GEC</td>
</tr>
<tr>
<td>Turret locking pin (shear)</td>
<td>0.82</td>
<td>GEC</td>
</tr>
<tr>
<td>Trolley X-restraint (shear)</td>
<td>0.53</td>
<td>GEC</td>
</tr>
</tbody>
</table>
These results were obtained applying the seismic response spectra for accelerations acting at the MHM runway level resulting from a DBE having an NPH of 0.35 g horizontally and 0.23 g vertically above 33.33 Hz. Similarly, the receiving crane confirmatory analysis documented in Calculation CSB-S-0031, Rail System Calculations, provided rail loads due to seismic loading on the rails. These results are contained in Table A4-7. Structural analyses were performed for the rails and rail frogs in accordance with the Manual of Steel Construction Ninth Edition (AISC 1989), and ACI 318-89, Building Code Requirements for Reinforced Concrete. The results of these analyses for the MHM rails and frogs under the above loading conditions are contained in CSB-S-0031. A summary of the demand/capacity ratios is provided in Table A4-8.

Load combinations for steel components are:

\[ 1.7A_s = D + L \pm Ed/F_{\mu} \]
\[ 1.7A_s = 0.85D \pm Ed/F_{\mu} \]

and load combinations for concrete are:

\[ U = D + L \pm Ed/F_{\mu} \]
\[ U = D \pm Ed/F_{\mu} \]

where

- \( A_s \) = allowable stress
- \( U \) = ultimate concrete stress
- \( D \) = dead load
- \( L \) = live load
- \( Ed \) = design basis earthquake load
- \( F_{\mu} \) = allowable seismic inelastic demand capacity ratio.

A4.3.10.5 Controls (Technical Safety Requirements). The MHM rails and rail frogs are design features and their safety function is managed by the configuration control process (see Section A5.6). The following assumption associated with the MHM rails and rail frogs requires TSRs to ensure performance of the safety function.

- Periodic surveillance of the rail clip hold-down bolts will verify that the bolts have sufficient torque as defined in AISC (1989). Verification of the rail frog's hold-down bolt torque must be performed each time the rail frogs are repositioned (see Chapter A5.0, Section A5.5.3.3) and before the MHM's seismic clamps are positioned over them to ensure consistency with the analysis results in CSB-S-0031.
<table>
<thead>
<tr>
<th>Load combinations</th>
<th>Dead load, kips/wheel</th>
<th>Live load, kips/wheel</th>
<th>EW seismic Z-direction, kips/wheel</th>
<th>NS seismic Z-direction, kips/wheel</th>
<th>NS seismic X-direction, kips/wheel</th>
<th>Vertical seismic, kips/wheel</th>
<th>Total P (Fz) vertical, kips/wheel</th>
<th>Total V (Fx) horizontal, kips/wheel</th>
</tr>
</thead>
<tbody>
<tr>
<td>Maximum down</td>
<td>60.41</td>
<td>52.4</td>
<td>104.4</td>
<td>46.8</td>
<td>81</td>
<td>55.63</td>
<td>223.622</td>
<td>81</td>
</tr>
<tr>
<td>Uplift 1</td>
<td>60.41</td>
<td>52.4</td>
<td>104.4</td>
<td>46.8</td>
<td>81</td>
<td>29.8</td>
<td>-49.1315</td>
<td>81</td>
</tr>
<tr>
<td>Uplift 1A</td>
<td>60.41</td>
<td>52.4</td>
<td>104.4</td>
<td>46.8</td>
<td>81</td>
<td>29.8</td>
<td>-40.07</td>
<td>81</td>
</tr>
<tr>
<td>Uplift 2</td>
<td>60.41</td>
<td>52.4</td>
<td>104.4</td>
<td>46.8</td>
<td>81</td>
<td>29.8</td>
<td>-83.6915</td>
<td>32.4</td>
</tr>
<tr>
<td>Uplift 2A</td>
<td>60.41</td>
<td>52.4</td>
<td>104.4</td>
<td>46.8</td>
<td>81</td>
<td>29.8</td>
<td>-74.63</td>
<td>32.4</td>
</tr>
</tbody>
</table>

**Unweighted End**

<table>
<thead>
<tr>
<th>Load combinations</th>
<th>Dead load, kips/wheel</th>
<th>Live load, kips/wheel</th>
<th>EW seismic Z-direction, kips/wheel</th>
<th>NS seismic Z-direction, kips/wheel</th>
<th>NS seismic X-direction, kips/wheel</th>
<th>Vertical seismic, kips/wheel</th>
<th>Total P (Fz) vertical, kips/wheel</th>
<th>Total V (Fx) horizontal, kips/wheel</th>
</tr>
</thead>
<tbody>
<tr>
<td>Maximum down</td>
<td>49</td>
<td>7.6</td>
<td>62.64</td>
<td>46.8</td>
<td>81</td>
<td>27.92</td>
<td>139.624</td>
<td>81</td>
</tr>
<tr>
<td>Uplift 1</td>
<td>49</td>
<td>7.6</td>
<td>62.64</td>
<td>46.8</td>
<td>81</td>
<td>24.17</td>
<td>-39.874</td>
<td>81</td>
</tr>
<tr>
<td>Uplift 1A</td>
<td>49</td>
<td>7.6</td>
<td>62.64</td>
<td>46.8</td>
<td>81</td>
<td>24.17</td>
<td>-32.524</td>
<td>81</td>
</tr>
<tr>
<td>Uplift 2</td>
<td>49</td>
<td>7.6</td>
<td>62.64</td>
<td>46.8</td>
<td>81</td>
<td>24.17</td>
<td>-49.378</td>
<td>32.4</td>
</tr>
<tr>
<td>Uplift 2A</td>
<td>49</td>
<td>7.6</td>
<td>62.64</td>
<td>46.8</td>
<td>81</td>
<td>24.17</td>
<td>-42.028</td>
<td>32.4</td>
</tr>
</tbody>
</table>

Dead load = bridge weight + shield weight + hoist weight.
1 kip = 1,000 lb.
P = force in Z-direction (Fz).
V = force in X-direction (Fx).

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### Table A4-8. Summary of Demand-to-Capacity Ratios for Multi-Canister Overpack Handling Machine Rails and Rail Frogs.

<table>
<thead>
<tr>
<th>Middle rail section</th>
<th>Description</th>
<th>D/C</th>
<th>End rail section</th>
<th>Description</th>
<th>D/C</th>
<th>Rail frog</th>
<th>Description</th>
<th>D/C</th>
</tr>
</thead>
<tbody>
<tr>
<td>Rail clips flexure</td>
<td>0.75</td>
<td></td>
<td>1/2 in. weld</td>
<td>0.76</td>
<td></td>
<td>1/2 in.</td>
<td>rail frog</td>
<td>0.45</td>
</tr>
<tr>
<td>1-1/4 in. anchor bolts</td>
<td>0.85</td>
<td></td>
<td>1-1/2 in. anchor bolts</td>
<td>0.85</td>
<td></td>
<td>1-1/2 in. anchor bolts</td>
<td>0.42</td>
<td></td>
</tr>
<tr>
<td>2 in. plate flexure</td>
<td>0.71</td>
<td></td>
<td>1-5/8 in. plate flexure</td>
<td>0.97</td>
<td></td>
<td>2 in. plate flexure</td>
<td>0.31</td>
<td></td>
</tr>
</tbody>
</table>

*D/C = demand/capacity.*

### A4.4 SAFETY-SIGNIFICANT STRUCTURES, SYSTEMS, AND COMPONENTS

The CSB safety-significant SSCs are described in Sections A4.4.1 through A4.4.21 and are summarized in Table A4-9. These items are designated safety significant in accordance with DOE-STD-3009-94, Letter 97-SFD-172 (Sellers 1997), and where applicable, ITS using the graded approach in HNF-SD-SNF-DB-003. Safety-significant SSCs provide design features and controls for safe handling of an MCO and for worker safety and also provide defense-in-depth safety features as defined in HNF-PRO-097. Where the design, construction, or procurement preceded HNF-PRO-704 implementation, waivers to safety-class designation of SSCs were documented. The safety-class designation would have been required due to their role in protecting safety-class SSCs, and from physical interaction. Previous procedures allowed a "3 over 1" designation for SSCs performing in that role. These SSCs are included herein as safety-significant SSCs.

The industry codes and standards used for the design, fabrication, and procurement of safety-significant SSCs are shown in Table A4-10. Safety-significant SSCs and services are procured from vendors whose quality assurance programs have been evaluated against the applicable quality assurance program requirement, such as ASME NQA-1-1994, *Quality Assurance Program Requirements for Nuclear Facilities*, or Title 10, *Code of Federal Regulations*, Part 830, “Nuclear Safety Management,” Section 120, “Quality Assurance Requirements” (10 CFR 830), and who have been placed on the Fluor Hanford Evaluated Suppliers List. In all cases, safety-significant SSCs are required to withstand NPH loads except for seismic criteria. Safety-significant SSCs may selectively be required to meet safety-class or industry standards for seismic events.
Table A4-9. Summary of Safety-Significant Structures, Systems, and Components. (5 sheets)

<table>
<thead>
<tr>
<th>Safety significant SSC (Chapter A4.X section)</th>
<th>Accident section (Chapter A3.X section)</th>
<th>Safety function</th>
<th>Functional requirements</th>
<th>Performance criteria (TSR applicability, Chapter A5.X section)</th>
<th>ITS category</th>
</tr>
</thead>
<tbody>
<tr>
<td>Operating area shelter and support area building foundation (A4.4.1)</td>
<td>A3.4.2.1</td>
<td>Protect operating area deck from structural failure of the operating area shelter</td>
<td>Withstand NPH and other design basis loading conditions</td>
<td>Configuration control (A5.6.5)</td>
<td>B (except support area building foundation, which is NA)</td>
</tr>
<tr>
<td>Standard storage tube plug (A4.4.2)</td>
<td>A3.3.2.3.3 worker safety</td>
<td>Protect worker from radiation dose in vault 1</td>
<td>Provides shielding from MCOs in vault 1 to meet ALARA goals</td>
<td>Verify tube plugs are in place (vault 1) (A5.5.3.5)</td>
<td>NA</td>
</tr>
<tr>
<td>Overpack storage tube plug (A4.4.3)</td>
<td>A3.3.2.3.3 worker safety</td>
<td>Protect worker from radiation dose in vault 1</td>
<td>Provides shielding from MCOs in vault 1 to meet ALARA goals</td>
<td>Verify tube plugs are in place (vault 1) (A5.5.3.5)</td>
<td>NA</td>
</tr>
<tr>
<td>MHM structural components; MCO hoist and grapple, and MHM interlocks P2, P6, and P21 (A4.4.4)</td>
<td>A3.4.2.1</td>
<td>Prevent damage to MCO or operating deck caused by structural failure of the MHM</td>
<td>Withstand all DBA structural loading conditions</td>
<td>MHM (A5.6.10, A5.3.2.10)</td>
<td>B (except interlocks, which are NA)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Provide safe handling of the MCO</td>
<td>Maintain critical geometry control</td>
<td>Operational checks and testing of related MHM interlocks (A5.5.1.1)</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Protect worker from radiation exposure from an open storage tube or from an MCO being handled by the MHM</td>
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</table>

Annex A - Canister Storage Building

March 2000
<table>
<thead>
<tr>
<th>Safety significant SSC (Chapter A4.X section)</th>
<th>Accident section (Chapter A3.X section)</th>
<th>Safety function</th>
<th>Functional requirements</th>
<th>Performance criteria (TSR applicability, Chapter A5.X section)</th>
<th>ITS category</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tube vent and purge cart (A4.4.5)</td>
<td>Non-DBA recovery action</td>
<td>Maintain confinement when connected to overpack storage tube plug</td>
<td>Provide for controlled venting of MCO gases</td>
<td>Recovery action, No TSRs (A5.3.2.12)</td>
<td>NA</td>
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<tr>
<td></td>
<td></td>
<td>Minimize releases of particulate from overpack storage tubes to the environment</td>
<td>Provide HEPA filtration of MCO vented gases</td>
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<tr>
<td>Receiving crane structure and hoist (A4.4.6)</td>
<td>A3.4.2.1</td>
<td>Provide safe handling of transportation cask and MCOs</td>
<td>Structural features of crane and hoisting equipment designed to national standards for lifting equipment</td>
<td>Verify correct cask lifting yoke in use for lift (A5.5.3.3, A5.6.9)</td>
<td>B</td>
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<tr>
<td></td>
<td></td>
<td>Withstand operating and seismic loading conditions</td>
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<td>NRC functional criteria (A5.3.2.9)</td>
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<tr>
<td>Cask receiving pit (A4.4.7)</td>
<td>A3.3.2.3.3</td>
<td>Protect worker from radiation dose</td>
<td>Provide shielding to meet ALARA goals</td>
<td>Configuration control (A5.6.5, A5.3.2.5)</td>
<td>NA</td>
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<tr>
<td>Transportation cask servicing system (flex connector, HEPA filter, piping between the filter and the cask, and PSV-102) (A4.4.8)</td>
<td>Non-DBA recovery action</td>
<td>Protect MCO from exceeding external design pressures. Maintain confinement when connected to the transportation cask Minimize releases of particulate from transportation cask Prevent buildup of flammable gases in transportation cask</td>
<td>Provide helium supply gas overpressurization protection Provide for controlled venting of gases from the transportation cask Provide HEPA filtration of gases Provide helium supply to transportation cask</td>
<td>Recovery action, no TSRs (A5.3.2.13)</td>
<td>NA</td>
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<td>Safety significant SSC (Chapter A4.X section)</td>
<td>Accident section (Chapter A3.X section)</td>
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<tr>
<td>MCO sampling system (The sampling/weld station MCO support structure, sample hood and HEPA filter, sampling piping confinement system, the exhaust system from the sample hood to the exhaust fan HEPA filter, sample hood flow indicator, center shield plate, and MCO valve operator) (A4.4.9)</td>
<td>A3.4.2.1</td>
<td>Maintain confinement when connected to an MCO</td>
<td>Pressure boundary designed for DBA and DBE conditions</td>
<td>Placement of shielding (A5.6.8, A5.6.11, A5.5.3.3)</td>
<td>NA</td>
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<tr>
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<td>A3.4.2.1</td>
<td>Minimize releases of particulate from MCO</td>
<td>Provide overpressurization protection, hood exhaust system, and HEPA filtration</td>
<td>Controls for sampling hood exhaust system (LCO 3.2.1)</td>
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<td>Prevent flammable mixture in MCO or in sampling equipment</td>
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<td>Flow rate for sampling hood exhaust system (A5.5.2.1)</td>
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<td>Protect MCO from NPH and load drop events</td>
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<td>Leak test piping and MCO valve operator (A5.5.2.2)</td>
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<td>Prevent equipment moving horizontally from damaging the sampling system</td>
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<td>Standard storage tube intermediate impact absorber (A4.4.10)</td>
<td>A3.4.4</td>
<td>Maintain confinement when connected to an MCO</td>
<td>Pressure boundary designed for DBA and DBE conditions</td>
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<td>Impact absorber in place (A5.5.3.4)</td>
<td>NA</td>
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<tr>
<td>Helium supply system rupture disk (A4.4.11)</td>
<td>A3.4.2.1</td>
<td>Protect bottom MCO from damage caused by dropping a second MCO into a storage tube</td>
<td>Absorb impact loading</td>
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<td>Impact absorber in place (A5.5.3.4)</td>
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<td>MHM fixed shielding system (A4.4.12)</td>
<td>A3.3.2.3.3</td>
<td>Protect worker from radiation exposure from the MHM</td>
<td>Provide shielding to meet ALARA goals</td>
<td>Configuration control (A5.6.8)</td>
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<tr>
<td>Transportation cask shielding (A4.4.13)</td>
<td>A3.3.2.3.3</td>
<td>Protect the facility operator from excessive radiation dose</td>
<td>Limit radiation field to 0.5 mrem/h at 30 cm</td>
<td>Conforms to design configuration (A5.6.1)</td>
<td>NA</td>
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<td>Safety significant SSC (Chapter A4.X section)</td>
<td>Accident section (Chapter A3.X section)</td>
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<td>MCO shield plug shielding (A4.4.14)</td>
<td>A3.3.2.3.3 worker safety</td>
<td>Protect facility operator from excessive radiation dose</td>
<td>Limit radiation field from the top of an MCO to an average of 2 mrem/h and not greater than 8 mrem/h</td>
<td>Conforms to design configuration (A5.6.2)</td>
<td>NA</td>
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<td>Standard interface guide ring funnel, storage tube lower flange, and bottom impact absorber (A4.4.15)</td>
<td>A3.4.2.1</td>
<td>Mitigate drop and impact of MCO</td>
<td>Prevent damage to an MCO</td>
<td>Configuration control (A5.6.6)</td>
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<td>Interface guide ring funnel and impact absorber in place (A5.5.3.3, A5.5.3.4)</td>
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<td>Overpack interface guide ring funnel, storage tube lower flange, and bottom impact absorber (A4.4.16)</td>
<td>A3.4.2.1</td>
<td>Mitigate drop and impact of MCO</td>
<td>Prevent damage to an MCO</td>
<td>Configuration control (A5.6.7)</td>
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<td>Interface guide ring funnel and impact absorber in place (A5.5.3.3, A5.5.3.4)</td>
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<tr>
<td>Cask lifting yoke (A4.4.17)</td>
<td>A3.4.2.1</td>
<td>Prevent lifting MCO to a height from which it would sustain unacceptable damage if dropped</td>
<td>Meet ANSI N14.6-1993* for design of special lifting devices Prevent crane from lifting cask-MCO with yoke attached to a height greater than 60 in. above floor</td>
<td>Verify correct cask lifting yoke in use for lift (A5.5.3.3) Cask lifting yoke length limits lift height of cask bottom to 60 in. (A5.6.15)</td>
<td>NA</td>
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<tr>
<td>MCO centering guide (A4.4.18)</td>
<td>A3.4.2.1</td>
<td>Prevent the MCO from exceeding the analyzed eccentric drop</td>
<td>Guide a dropped MCO into a storage tube</td>
<td>Ensure placement of MCO centering guide in the MFM (A5.5.3.3)</td>
<td>NA</td>
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<tr>
<td>Cask receiving impact absorber (A4.4.19)</td>
<td>A3.4.2.1</td>
<td>Protect cask-MCO dropped by receiving crane into cask receiving pit from unacceptable damage</td>
<td>Absorb energy from dropped cask-MCO to within design limits</td>
<td>Cask receiving impact absorber in place (A5.5.3.4)</td>
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</tr>
<tr>
<td>Safety significant SSC (Chapter A4.X section)</td>
<td>Accident section (Chapter A3.X section)</td>
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<tr>
<td>Sampling/weld station impact absorber and shield halves (A4.4.20)</td>
<td>A3.4.2.1</td>
<td>Protect MCO dropped by crane into sampling/weld station from unacceptable damage</td>
<td>Absorb energy from dropped MCO to within design limits</td>
<td>Sampling/weld station impact absorber and shield halves in place (A5.5.3.3, A5.5.3.4)</td>
<td>NA</td>
</tr>
<tr>
<td></td>
<td>A3.3.2.3.3</td>
<td>Protect worker from radiation dose</td>
<td>Provides shielding to meet ALARA goals</td>
<td>Configuration control (A5.6.8)</td>
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<tr>
<td>Shield hatch and MCO guide assembly (A4.4.21)</td>
<td>A4.3.2.1</td>
<td>Prevent damage to MCO caused by structural failure</td>
<td>Withstand all DBA structural loading conditions</td>
<td>Shield hatch and MCO guide assembly in place (A5.5.3.3)</td>
<td>NA</td>
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<tr>
<td></td>
<td>A3.3.2.3.3</td>
<td>Prevent rearrangement of MCO internals</td>
<td></td>
<td>Configuration control (A5.6.8, A5.6.12)</td>
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</tr>
</tbody>
</table>


ALARA = as low as reasonably achievable.
DBA = design basis accident.
DBE = design basis earthquake.
HEPA = high-efficiency particulate air (filter).
ITS = important to safety.
LCO = Limiting Conditions for Operation.
MCO = multi-canister overpack.
MHD = multi-canister overpack handling machine.
NA = not applicable (this SSC does not have an ITS category).
NPH = natural phenomena hazard.
NRC = U.S. Nuclear Regulatory Commission.
SSC = structure, system, or component.
TSR = technical safety requirement.
### HNF-3553 REV 0
#### Annex A — Canister Storage Building

<table>
<thead>
<tr>
<th>Component Tag Number</th>
<th>Component Description</th>
<th>Safety Function</th>
<th>Design and Fabrication Code/Standard</th>
</tr>
</thead>
<tbody>
<tr>
<td>NA</td>
<td>Operating area</td>
<td>Protect from structural failure of the operating shelter and structural failure of the support area building</td>
<td>AISC (1999), UBC (1994), NFPA 10, NFPA 780, ASTM A27/A27M, ASTM A53</td>
</tr>
<tr>
<td>PL-001</td>
<td>Standard storage area building</td>
<td>Protect workers from radiation</td>
<td>ASME B30.20-1993, ASTM A27/A27M, ASTM A53</td>
</tr>
<tr>
<td>PL-002</td>
<td>Overpack storage area building</td>
<td>Protect workers from radiation</td>
<td>ASME B30.20-1993, ASTM A27/A27M, ASTM A53</td>
</tr>
<tr>
<td>CRN-004, ESR-001</td>
<td>MBM structural components, MBM boxes and grapples, and MBM interlocks</td>
<td>Perform recovery operations involving MCOs</td>
<td>ASME B30.1995, CMAA 70-94, ANSI 14-4.6-1993</td>
</tr>
<tr>
<td>FH-2</td>
<td>Cask receiving pit</td>
<td>Protect workers from radiation</td>
<td>10 CFR 855, ASTM A36/A36M, ASTM A490</td>
</tr>
</tbody>
</table>

**Table A4-10, Safety-Significant Canister Storage Buildings, Structures, and Components (5 sheets)**

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<table>
<thead>
<tr>
<th>Component description</th>
<th>Component tag number</th>
<th>Safety function</th>
<th>Design and fabrication code/standard</th>
<th>Rationale</th>
</tr>
</thead>
<tbody>
<tr>
<td>Standard storage tube intermediate impact absorber (A4.4.10)</td>
<td>IMP-003</td>
<td>Protect bottom MCO and top MCO</td>
<td>ASTM A36/A36M, ASTM A312/A312M</td>
<td>B</td>
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<tr>
<td>Helium supply system rupture disk (PSE-1) (A4.4.11)</td>
<td>PSE-1</td>
<td>Prevent damage to an MCO</td>
<td>ASME B16.34</td>
<td>B</td>
</tr>
<tr>
<td>MHM fixed shielding system (A4.4.12)</td>
<td>CRN-004, RSC-001</td>
<td>Protect workers from radiation</td>
<td>10 CFR 20.1601, 10 CFR 20.1301, NRC Regulatory Guide 8.8</td>
<td>B</td>
</tr>
<tr>
<td>Transportation cask shielding (A4.4.13)</td>
<td></td>
<td>Protect workers from radiation</td>
<td>10 CFR 20.1601, 10 CFR 20.1301</td>
<td>B</td>
</tr>
<tr>
<td>MCO shield plug shielding (A4.4.14)</td>
<td></td>
<td>Protect workers from radiation</td>
<td>10 CFR 20.1601, 10 CFR 20.1301</td>
<td>B</td>
</tr>
<tr>
<td>Standard interface guide ring funnel, storage tube lower flanges and bottom impact absorbers (A4.4.15)</td>
<td>PL-001, IMP-001</td>
<td>Prevent damage to an MCO</td>
<td>ASTM A36/A36M, ASTM A513, ANSI/AISC N690-94</td>
<td>B</td>
</tr>
</tbody>
</table>
### Table A4-10  Safety-Significant Canister Storage Building Systems, Structures, and Components  

<table>
<thead>
<tr>
<th>Component description</th>
<th>Component tag number</th>
<th>Safety function</th>
<th>Design and fabrication code/standard</th>
<th>Rationale</th>
</tr>
</thead>
<tbody>
<tr>
<td>Overpack interface guide ring funnel, storage tube lower flanges and bottom impact absorbers (A4.4.16)</td>
<td>PI-001, IMP-001</td>
<td>Prevent damage to an MCO</td>
<td>ASTM A36/A36M, ASTM A513, ANSI/AISC N690-94</td>
<td>B</td>
</tr>
<tr>
<td>Cask lifting yoke (A4.4.17)</td>
<td>Tool 98</td>
<td>Prevent gas releases from a dropped MCO</td>
<td>ASTM A36/A36M, ANSI N14.6-1993</td>
<td>B</td>
</tr>
<tr>
<td>MCO centering guide (A4.4.18)</td>
<td>IMP-004</td>
<td>Protect the MCO from damage</td>
<td>ASTM A36/A36M</td>
<td>B</td>
</tr>
<tr>
<td>Cask receiving impact absorber (A4.4.19)</td>
<td>IMP-005, RSE-007, RSE-008</td>
<td>Protect the MCO from damage</td>
<td>ASTM A36/A36M, ASTM A53, ASTM 513, ASTM A193/A193M, ANSI/AISC N690-94</td>
<td>B</td>
</tr>
<tr>
<td>Sampling/weld station impact absorber and shield halves (A4.4.20)</td>
<td>RSE-004</td>
<td>Protect the MCO from damage</td>
<td>ASTM A36/A36M, ASTM 513</td>
<td>B</td>
</tr>
</tbody>
</table>

Table A4-10. Safety-Significant Canister Storage Building Systems, Structures, and Components. (5 sheets)

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<thead>
<tr>
<th>Component description</th>
<th>Component tag number</th>
<th>Safety function</th>
<th>Design and fabrication code/standard</th>
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<td>Atlanta, Georgia.</td>
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<td>Engineers, New York, New York.</td>
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<tr>
<td>Component tag number</td>
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<td>NFPA 255, 1990, Standard Method of Test of Surface Burning Characteristics of Building Materials</td>
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A4.4.1 Operating Area Shelter and Support Area Building Foundation

A4.4.1.1 Safety Function. The safety function of the operating area shelter is to protect the operating area deck from structural failure of the operating area shelter. The safety function of the support area building foundation is to protect the operating area shelter from structural failure of the support area building. The operating area shelter is designed so that its failure will not result in failure of the safety-class deck below. Therefore, the operating area shelter is designed for NPH events listed in Table A1-1 including DBE and tornado wind loads. Because of the proximity of the support area building to the operating area shelter, the main frame of the support area building also is subjected to design basis loading for DBE and tornado wind. Tornado and DBE loads are transmitted to the support area building foundation. The support area building is designed to not fail and cause failure of any CSB SSC. HNF-PRO-704 would have required that the support building and operating area shelter be designated safety class after the completion of construction. The project prepared and obtained a waiver to HNF-PRO-704, Table 5, Note 4 (Waiver). The operating area shelter, including the rolling shield gates, and support area building foundation are designated as safety significant in accordance with Letter 97-SFD-172 (Sellers 1997) and as identified in Section A3.4.2.1.

A4.4.1.2 System Description. The operating area shelter is a steel-framed enclosure that covers the operating area deck, the load-in/load-out area on the north, and the sampling/weld area on the south. The building structural system is steel ordinary moment resisting frame (columns/truss) in the east-west direction, and steel concentric diagonal braced frame in the north-south direction (see Section A2.4.4.2 and Figure A2-1). The building is approximately 230 ft long, 136 ft wide, and 55 ft high and is supported on a 3-ft-high concrete curb that surrounds the operating area deck. The structural system consists of load-carrying roof trusses spanning east–west and supported by columns. Seismic and tornado lateral forces are resisted through truss-column frame interaction in the east–west direction and through a vertical bracing system along the east and west walls in the north–south direction. Rolling shield gates at the north trailer vestibule and south load-in/staging area (see Figure A2-18) protect the telescoping doors components from wind forces, including tornado missiles, and serve as security barriers. The telescoping doors are general service. However, without the protection provided by the gates, the telescoping doors would fail and permit missiles inside the operating shelter. For this reason, the rolling shield gates are classified safety significant and designed to withstand the same loads from Table A1-1 as the operating area shelter.

The support area building is an extension to the north of the operating deck and the MCO load-in/load-out area. The support area building’s foundation dimensions are 150 ft by 54 ft, 3 in. The foundation system consists of reinforced concrete spread footings. The support area building is a steel-framed structure approximately 150 ft by 54 ft by 16 ft high enclosed by insulated metal siding and roof deck. A heating, ventilation, and air conditioning (HVAC) exhaust stack projects upwards from the support building roof (see Section A2.4.4.3 and Figure A2-1).

A4.4.1.3 Functional Requirements. The operating area shelter must not be damaged by the DBE or other NPH loads listed in Table A1-1 so that safety-class items under the shelter (e.g., the...
operating area deck) can maintain their safety functions. For continuity of design and to eliminate interaction effects, the support area building must not be damaged by the DBE and other NPH loads listed in Table A1-1 to prevent damage to the nearby operating area shelter. The standing seam architectural cladding is able to sustain the 90 mi/h design basis wind load. The support building's exhaust stack is required to withstand the design basis tornado and DBE loads listed in Table A1-1. The support area building foundation is designed for the following design basis loading conditions:

- Roof dead load
- Roof live load (20 lb/ft²)
- Floor live load (100 lb/ft²)
- Roof snow load based on ground-level snow load plus drift
- Siding dead load
- Equipment weight (consideration for vibration and/or overturning accounted for in the foundation design)
- DBE seismic for free-field response spectra for interaction with operating deck
- Tornado loads (wind 200 mi/h)
- Extreme wind (90 mi/h, including 2 in. by 4 in. wood plank, 15 lb missile at 50 mi/h).

A4.4.1.4 System Evaluation. A three-dimensional model of the operating area shelter was subjected to the basic static load cases, including the 200 mi/h tornado load with atmospheric pressure change, and to response spectra dynamic analysis. The horizontal response applied was developed using a vault model, with the control motion being applied at the competent soil layer, and the soil properties were varied using one-half mean, mean, and two times mean values. For vertical response, the building was subjected to two-thirds of the free-field control motion. Loads were combined and the demand-to-capacity ratios for the structure were developed. One weather-sealed penetration (roof hatch) through the roof (5 ft, 3 in. by 5 ft, 3 in. clear opening), installed between the existing purlins and roof beams, provides for outside crane access to the operating area. Installing the roof hatch between the existing roof support structures has minimal effect on the calculated demand-to-capacity ratios.

\( F_\mu \) values > 1.0 per DOE-STD-1020-94, Table 2-4, were used for member evaluation in combination with snow and other loads, but the connection designs are based on \( F_\mu = 1.0 \). The recommendations of the Uniform Building Code (UBC 1994) pertaining to good seismic detailing were used in the design of the CSB operating area shelter and support building. Analysis for 25% of the design basis snow load of 20 lb/ft², with the 20 lb/ft² roof live load and 17 lb/ft² roof dead
load combined with DBE for the controlling members was performed with appropriate $F_p$ factors
greater than 1.00 as allowed by DOE-STD-1020-94, Table 2-4, based on governing tensile, shear,
or compressive loads. $F_p$ values of 1.00 were used in calculations CSB-S-0029 and CSB-S-0048,
which resulted in demand capacity ratios less than 1.0. A post-construction check of the
operating shelter roof dead load estimated that the actual dead load is 15.8 lb/ft² (CSB-S-0048,
Supplement H). This post-construction analysis confirms that the 17 lb/ft² used in the DBE
analysis is conservative.

The demand-to-capacity ratios calculated include the tornado and seismic loads. The
combination of dead-plus-live load with tornado wind results in demand-to-capacity ratios for the
operating area shelter that are lower than dead load plus tornado wind. Calculated
demand-to-capacity ratios of load combinations for dead-plus-live load with tornado wind are
acceptable. Demand-to-capacity ratios of dead load plus earthquake (free field and at top of
competent layer with $C_v = 1.0$), dead load plus tornado wind, and dead load plus snow with
ashfall loads also are acceptable.

Seismic and tornado wind analysis of the HVAC exhaust was conservatively carried out
for the design basis loads in Table A1-1 using RISA-3D per the load combinations required by
AISC (1989). The results indicate that tornado wind with dead load ($1.05D + W_t$) governs the
design demand/capacity ratio of 0.84 for flexure and is the most limiting.

Seismic motions of the operating area shelter and intake structure were evaluated in
CSB-S-0048, Supplement H (see Section A4.3.6.4). Motions up to 5.0 in. are predicted across
the 4.75-5.375 in. gap (see Table A4-4). Damage to the building's roof line from the seismic
movement of the exhaust stack and operating area building is predicted for the eave and gutter
and the architectural standing seam roofing; some buckling of the structural roof deck is also
expected. These items are either sacrificial or easily repaired and pose no restrictions to
continued operation following a seismic event. No energy-absorbent materials are warranted.
The standing seam architectural cladding is required only to sustain the 90-mi/h design basis wind
load, not the tornado wind load.

The wind missile specified in Letter FRP-112 (Bedell 1996a) (15 lb, 2 in. by 4 in. board
with a horizontal velocity of 73.5 ft/s and a maximum height of 50 ft) is consistent with
DOE-STD-1020-94 and HNF-PRO-097. The steel barrier thickness required to prevent
penetration from this missile is one sheet of 13-gauge or two sheets of 18-gauge material
(Bedell 1996a, Section 10B, page 26). The siding provided for the operating shelter is two sheets
of 18-gauge material.

The siding and structural framing for the operating shelter were evaluated for impact from
the wind-generated missile in Calculation CSB-S-0010, Operating Shelter Design/Analysis,
Sheets 107 through 109. The siding and girts (the lightest framing members) were evaluated
directly and found to be adequate for the missile impact. Based on the ratio of the missile mass to
the effective mass of the main framing members, the effect of the missile loads on the main
framing members was deemed to be insignificant. Subsequent to this analysis, the main framing
members for the operating area shelter were increased in size because of the imposition of the
tornado loading; therefore, no further consideration of wind-generated missiles was necessary.

The support area building structure is designed and analyzed as a moment frame in the
north-south direction and as a braced frame in the east-west direction. The steel framing, roof
deck, and siding are designed to resist wind pressures from a 200 mi/h tornado, and the primary
framing also is designed for a DBE. The calculated demand-to-capacity ratios of the primary
structural elements are acceptable. The limiting loading condition for the columns is the
combination of dead load and snow load with tornado wind \(1.05D+S+W_t\).

A4.4.1.5 Controls (Technical Safety Requirements). The operating area shelter and the
support area building foundation are design features (see Section A5.6), and their safety function
is managed by the configuration control process.

A4.4.2 Standard Storage Tube Plug

A4.4.2.1 Safety Function. The standard storage tube plugs are classified safety significant in
accordance with DOE-STD-3009-94 and provide the safety function to protect workers from
radiation dose in vault I (see Chapter A3.0, Section A3.3.2.3.3).

A4.4.2.2 System Description. The standard storage tube plugs are a composite cast steel upper
section with a carbon steel, cast-in-place concrete-filled cylindrical lower section. The composite
cast steel upper section meets the specifications in ASTM A27/A27M, Standard Specification for
Steel Castings; Casting for General Application, and the carbon steel in the lower section meets
the specifications in ASTM A36/A36M for rolled plate or pipe. The weight of a standard storage
tube plug is approximately 5,300 lb. A 4-in.-diameter lifting pintle shown on Figure A2-14,
Sheet 1, is used for removal and installation of the standard storage tube plug. The carbon steel
pintle material conforms to ASTM A108, Standard Specification for Steel Bars, Carbon, Cold-
Finished, Standard Quality, Grade AISI 1040 or 1045. The three jaws of the MHM tube plug
hoist and grapple close and lock on the 2-in.-diameter machined “neck” of the pintle during
movement of the standard storage tube plug. The standard storage tube plug fiber filter media
(90% efficiency) are contained within metal canisters. The breather filter with a quick-connect
coupler is shown on Figure A2-14. The filter media housing and quick connect coupler are
attached to a quick connect nipple which screws into an ASTM A106, Standard Specification for
Seamless Carbon Steel Pipe for High Temperature Service, pipe coupling. The coupling is
screwed onto an ASTM A53, Standard Specification for Pipe, Steel, Black and Hot Dipped
Zinc-Coated (Galvanized) Welded and Seamless, pipe nipple. The other opening in the standard
tube plug assembly has a quick connect screwed into a ball valve which is screwed on the
ASTM A53 pipe nipple (Figure A2-14, sheet 1). Pressure in the standard storage tube is
maintained at atmospheric pressure as air from the operating area flows in and out of the breather
filter. The tube temperature is expected to range from 70 °F to 150 °F. The standard storage
tube plug has one O-ring seal.
A4.4.2.3 Functional Requirements. The requirement for shielding protection of the facility worker by the standard storage tube plugs is governed by the ALARA (as low as reasonably achievable) goal and related principles and practices for radiation protection. The plug's concrete-filled barrel (standard structural un-reinforced) and heavy cast steel top composition are provided to achieve this goal.

A4.4.2.4 System Evaluation. The standard storage tube plugs provide radiation shielding for personnel working above the standard storage tubes that contain MCOs. Calculations for standard storage tube plugs (CSB-SH-2002) estimate the dose at the surface of the operating deck to be about 0.2 mrem/h (2.0 × 10^-3 mSv/h). This value excludes any shielding contribution from the standard tube plug cover to provide shielding. An extremity dose of about 0.4 mrem/h (4.0 × 10^-3 mSv/h) is estimated for surveillance operations on the storage tube plugs below the tube plug covers. The standard storage tube plug design (Figure A2-8) meets the requirements for personnel shielding safety in accordance with HNF-SD-SNF-DB-003 (Items 20 and 23) and minimizes the spread of contamination. Standard storage tube plugs have a seal to provide a physical barrier and a breather filter to contain particulate and maintain atmospheric pressure within the standard storage tube. Standard tube plugs are filled with concrete at the job site. Typical concrete mix densities range from 2.33 g/cm³ to 2.40 g/cm³. Concrete density used in the shielding analysis was 2.24 g/cm³.

The initial stress analysis documented in Calculation CSB-RM-0002, Plug Lifting Lug Stress, was performed to ASME B30.20-1993 safety factors of 3 on yield and 5 on tensile strength, assuming a test weight of 1.5 times the all steel overpack tube plug weight of 12,300 lb. This is 2.3 times the estimated 5,300 lb. weight of a composite steel-concrete standard storage tube plug. The analysis examined the critical areas for the plug load path; the plug female threads and the plug lifting attachment top transition from 4 in. to 2 in. The analysis determined safety factors of 4.2 and 6.5, respectively, for the top transition 4 in. lifting attachment point. The stress analysis demonstrates that the design of the pintle exceeds the yield and tensile safety factor requirements of ASME B30.20-1993.

A4.4.2.5 Controls (Technical Safety Requirements). The following assumptions associated with the standard storage tube plugs require TSRs to ensure performance of the safety function.

- Tube plugs are in place over every storage tube unless that tube is being accessed by the MHM.
- MHM safety features ensure that the MHM will not leave a storage tube location before replacing the tube plug. Operability of interlock P2 is discussed under Limiting Condition for Operation (LCO) 3.1.1 in Chapter A5.0.
A4.4.3 Overpack Storage Tube Plug

A4.4.3.1 Safety Function. Overpack storage tube plugs are designated safety significant because of their safety function to protect workers from radiation dose in vault 1.

A4.4.3.2 System Description. Section A2.4.3 contains a description of the overpack storage tube assemblies including the overpack storage tube plug and the overpack tube plug covers. These components are shown on Figures A2-14 and A2-15. The plugs are a composite cast steel upper section (ASTM A27/A27M) with a carbon steel (ASTM A36/A36M), cast-in-place concrete-filled cylindrical lower section. The concrete has a compressive strength of 3,000 lb/in² in accordance with ASTM C94, Standard Specification for Ready Mixed Concrete, and ACI 301, Specification for Structural Concrete. The weight of an overpack storage tube plug is approximately 6,200 lb. A 4-in.-diameter lifting pintle shown on Figure A2-14, sheet 2, is used for removal and installation of the overpack tube plug assembly. The carbon steel pintle material conforms to ASTM A108, Grade AISI 1040 or 1045. The three jaws of the MHM tube plug hoist and grapple close and lock on the 2-in.-diameter machined “neck” of the pintle during movement of the overpack storage tube plug. A pressure gauge connected to a quick connect coupler is shown on Figure A2-14, sheet 2. The quick-connect coupler is attached to a quick-connect nipple, which screws into an ASTM A106 pipe coupling, which is screwed on an ASTM A53 pipe nipple. Another opening in the overpack tube plug assembly has a quick connect screwed into a ball valve which is screwed on an ASTM A53 pipe nipple. The overpack storage tube plug has two O-ring seals in the 45° mating surface. The double O-ring seal pressure check connection assembly is a quick connect screwed on a 1/4 in. ASME SA213 (ASTM A213/A213M) 304 stainless steel tube, which has been welded to a side opening drilled through the steel shield and opening between the O-ring seals. Pressure in the overpack storage tube will be approximately 7 lb/in² gauge during MCO storage to prevent air from entering and mixing with gases potentially emitted from an off-normal MCO. A spider lock-down device holds the overpack storage tube plug in place (Figure A2-14). The tube temperature is expected to range from 70 °F to 150 °F. Other than sharing the same shielding approach of a composite steel with concrete-filled barrel plug, overpack tube plugs differ from the standard tube plugs in size, configuration, and function. The pressure-retaining function of the overpack storage tubes requires that the overpack tube plugs be provided with features that interface with the plug hold-down bolts and the spider lock down mechanism. Redundant seals and no overpressurization relief ensure containment of the inert gas and confinement of any released MCO gases (see Section A4.3.5 and Figure A2-8).

A4.4.3.3 Functional Requirements. Overpack storage tube plugs provide shielding for the operators from the SNF contained in the MCOs in vault 1 and are in compliance with Items 20 and 23 of HNF-SD-SNF-DB-003. Standard structural un-reinforced concrete and heavy cast steel top composition are provided to achieved this goal. Shielding analyses assume a concrete density of 2.24 g/cm³.

A4.4.3.4 System Evaluation. HNF-PRO-704, Table 2, lists design codes and standards for design of SSCs based on their safety designation. Process equipment (e.g., vessels and tanks)
designated safety significant can be designed in accordance with the *Boiler and Pressure Vessel Code*, Section VIII (ASME 1995). Item 10 of HNF-SD-SNF-DB-003 requires that NRC Regulatory Guide 1.26 be used in assigning the appropriate code class to systems and components designed to the *Boiler and Pressure Vessel Code*, Section III (ASME 1995). The Category C quality level from NRC Regulatory Guide 1.26 has been assigned to the overpack storage tube assemblies because of the safety-class mitigative features described in Section A4.3.5. The overpack storage tube assemblies, including the overpack storage tube plug, are designed and fabricated in accordance with the *Boiler and Pressure Vessel Code*, Section III, Class NC (ASME 1995). For continuity of design, the bellows (Figure A2-8, sheet 2), although outside of the pressure boundary for the overpack storage tube assembly, also is designed and fabricated to the requirements of *Boiler and Pressure Vessel Code*, Section III, Class NC.

The overpack storage tube plugs provide shielding for the facility operators from the SNF contained in the MCOs below the deck in vault 1. Results of calculations for standard storage tube plugs estimate the dose at the surface of the operating deck to be about 0.2 mrem/h (2.0 \times 10^{-3} \text{ mSv/h}). An extremity dose of about 0.4 mrem/h (4.0 \times 10^{-3} \text{ mSv/h}) is estimated for surveillance operations on the overpack storage tube plugs below the deck (see CSB-SH-2002 and Section A4.4.2.4 above). The overpack storage tube plug design and the spider hold-down device meet the requirements for personnel shielding safety in accordance with HNF-SD-SNF-DB-003 (Items 20 and 23).

A stress analysis (CSB-RM-0002) was performed to ASME B30.20-1993 safety factors of 3 on yield and 5 on tensile strength, assuming a test weight of 1.5 times the all steel overpack tube plug weight of 12,300 lb (2 times the estimated 6,200 lb. weight of a composite steel-concrete overpack storage tube plug). The analysis examined the critical areas for the plug load path: the plug female threads and the plug lifting attachment top transition from 4 in. to 2 in. The analysis determined safety factors of 5.2 and 8.1 (respectively, yield and tensile) for the female plug threads and safety factors of 4.2 and 6.5, respectively, for the top transition 4 in. lifting attachment point. The stress analysis demonstrates that the design of the pintle exceeds the yield and tensile safety factor requirements of ASME B30.20-1993.

**A4.4.3.5 Controls (Technical Safety Requirements)**. The following assumptions associated with the overpack storage tube plugs, overpack tube plug covers, and overpack tube deck embed covers require TSRs to ensure performance of the safety function.

- Overpack tube plugs are in place over every overpack storage tube unless that tube is being accessed by the MHM.

- MHM safety features ensure that the MHM will not leave an overpack storage tube location before replacing the overpack tube plug. This interlock (P2) is described in Sections A4.4.4 and A4.4.12 and operability is discussed under LCO 3.1.1 in Chapter A5.0.
A4.4.4 Multi-Canister Overpack Handling Machine Structural Components; Multi-Canister Overpack Hoist and Grapple; and Multi-Canister Overpack Handling Machine Interlocks P2, P6, and P21

A4.4.4.1 Safety Function. Structural failure of the MHM could cause damage to the safety-class MCO or to the safety-class CSB operating deck. The safety function of these SSCs is to prevent damage to an MCO or the operating deck caused by structural failure of the MHM and to provide safe handling of the MCOs. The MHM is structurally classified as safety significant in accordance with Letter 97-SFD-172 (Sellers 1997) and DOE-STD-3009-94, and consistent with waiver 1 to HNF-PRO-704. The MHM MCO grapple and hoist assembly also is classified safety significant (Williams 1998a) and ITS Category B in accordance with the graded approach in HNF-SD-SNF-DB-003. The safety functions of the MHM structural components and MCO hoisting components are to provide a robust system to safely handle the MCO, to reduce the frequency and risk that the MCO will be dropped from or damaged by the MHM, and to reduce the frequency and risk that the operating deck will be damaged by the MHM during normal operations and during the DBE.

MHM operation is governed by a system of electrical interlocks. These interlocks provide permissives that allow the operator to perform MHM operations (e.g., actuate drives or motors) based on signals from a large number of sensors and switches. Interlocks P2, P6, and P21 have been designated safety significant for their safety function of protecting the MCO from drop and shear accidents and for providing worker radiation protection. In addition, MHM interlocks P3, P5, P8, P9, P26, P57, P61, P62, P63, P65, P66, P80, and P85 have been designated as defense-in-depth features for their safety function of providing additional design features and controls to prevent MCO drop and shear accidents.

A4.4.4.2 System Description. The MHM primarily consists of the bridge, trolley, and turret and turntable assemblies. The bridge is comprised primarily of bridge girders, wheel trucks, and a related drive system. The bridge is comprised primarily of a central deck, two outer box beams that carry the wheel and bearing assemblies, and a drive system. The turret and turntable assembly is comprised primarily of the MCO cavity, the tube plug cavity, the camera cavity, the shield skirt, two large slewing bearings, and a nose unit. The MCO hoist and grapple are located on the upper turret above and within the MCO cavity and consist of a drum and external drive assembly, an upper hoist assembly, and the grapple.

The grapple is a self-centering, self-aligning, six-jaw grapple designed to engage and lift, and lower and release MCOs and impact absorbers. The grapple configuration is designed to ensure that all the jaws can achieve the fully closed position without contacting the MCO. Each of the six jaws is operated via two pairs of pinned linkage arms. The jaws are mounted via pivot pins at one end of the linkage arms while the other end of the arms are attached to the jaw operating weight by a roller and bearing arrangement. The operating weight operates vertically with respect to the jaw carriers. When the jaws are to be opened, pressurized air is admitted to the cylinder to extend the piston, allowing the operating weight to lower under its own weight and open the jaws.
The crane’s bridge girders are plate box sections (6 ft wide by 8 ft high) with all-welded construction and stiffening diaphragms at proper intervals. The girders have plates for fastening to equalizing sills with large fitted bolts. Rails for trolley travel are fitted to the top of the girders and secured by bolted rail clips. The crane is carried on box sections, all-welded, pin-equalized trucks that are diaphragmed and plated for attaching to sills. Trucks are in-line machine bored with interface surfaces machined for accurate bolting of renewable wheel bearing cartridges. The bridge drive has two motors located at the trucks.

The trolley rides on the crane bridge girder rails and supports the MHM turret assembly, enabling it to traverse the east–west direction along the girders. The main trolley deck surrounds the turret and has the MHM operator station and the turret rotation drive and lock assembly mounted to it. The outer wheel beams are welded box sections with machined seatings formed at each end to receive the rail wheel bearing housings. All load points are stiffened with internal diaphragms. The trolley drive consists of a motor, brake, reduction gearbox, and twin cardan shaft driving two of the four trolley wheels.

The turret and turntable assembly is designed to ASME NOG-1-1995, Type I, crane requirements for permissible materials and stresses. The expected stresses for the turret and turntable assembly have been demonstrated to be within code allowable values. The turret consists of a rotating upper assembly and a nonrotating bottom nose unit. The upper turret assembly is supported from the trolley by a slewing bearing and the turntable. The upper turret is bolted to the turntable with a segmented packer at the interface that helps to provide a natural convection air passage for the MCO cavity. The nose unit consists of the lower turntable and retractable shield skirt. The nose unit is suspended from the underside of the upper turret by a slewing bearing. The nose unit is tied to the trolley via the base torsion linkage. The upper turret can be rotated relative to the trolley and the nose unit. The nose unit has one cavity or throughport that is eccentric to the center of the turntable slewing bearing. The upper turret has three cavity positions — MCO cavity, tube plug cavity, and television camera — any one of which may be placed over the nose throughport by rotating the upper turret to the proper location. The retractable shield skirt is mounted concentrically with and suspended from the nose unit and can be lowered to contact the deck to complete the MHM shielding during MCO hoisting or storage tube operations.

The MCO hoist is designed to CMAA 70-94, Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes, class D, for permissible materials and stresses. A failure modes and effects analysis of the hoist and grapple system has been completed consistent with MIL-STD-1629A, Procedures for Performing a Failure Modes, Effects and Criticality Analysis. The results of the analysis are provided in ESL/R(97)36, Failure Modes and Effects Analysis of Hanford MCO Hoist and Grapple. A stress analysis of the load path components for the MHM MCO grapple and hoist assembly is documented in ESL/R(96)099, Multi-canister Overpack Handling Machine Turret Design Calculations. The analysis confirms that the normal and seismic loads can be lifted in accordance with ASME NOG-1-1995 requirements and the Grapple Design Code ANSI N14.6-1993. These two references confirm that the MCO grapple and hoist are single-failure proof for lifting. The load
The load path components for the MCO grapple include the jaws, jaw pins, grapple body, link arms, pins, bolts and nuts, and flange ring shown on Figure A2-38, sheet 1. The load path components for the hoist assembly (Figure A2-38, sheet 2) include the load block, pulleys and pins, rope, balance beam, hoist support enclosure, main beam, pulley brackets, clevis, trunnions, and connectors. The hoist drum and external drive assembly are bolted to the top of the upper turret shield body, and the hoist upper assembly is bolted to the top of the drum structure. The hoist design uses a dual rope system with a twin-grooved drum (Figure A2-38, sheet 2). The two rope grooves in the drum are opposite-handed, helical grooves to ensure that both rope lengths remain identical throughout the MCO lift. The hoist drum is driven by a motor and primary drive train with a secondary drive train connected to the opposite end of the drum. An electric shoe brake provides the main load-holding brake for the drum. The secondary drive incorporates a second shoe brake and ensures that no credible single failure will result in the loss of ability to stop and hold the MCO load. The upper hoist assembly contains the reeving system for the two hoist ropes and the load block for attaching the grapple to the hoist system (Figure A2-38, sheet 2). The two hoist ropes are 0.75-in.-diameter, nonrotating ropes, each with a minimum breaking strength of 39 tons. The ropes travel from the drum up to twin head pulleys, down to the load block, and back up to the balancing beam where they are securely fixed. Guards on all pulleys ensure that the ropes remain within the pulley grooves. The load block clevis is mounted to the top of the MCO grapple lifting body and is suspended from the hoist ropes via two pulleys that are mounted such that either rope can perform a vertical lift if the other rope should fail. In the event that one rope should fail, the load on the remaining rope will overbalance the balancing beam, and damping cylinders ensure that the overbalancing motion will not lead to a shock loading on the rope and hoist and grapple mechanism that exceed allowable values from ASME NOG-1-1995, Section NOG-5425.

The MCO grapple is designed as a special lifting device for critical loads in accordance with Section 7 of ANSI N14.6-1993 for permissible materials, lower stresses, and higher proof load requirements. This results in high safety factors for the rated load. The expected stresses for the MCO grapple have been demonstrated to be within code allowable values. The MCO grapple is connected to the main hoist load block by the grapple lifting body, which lifts the grapple through the top plate. The grapple is a self-centering, self-aligning, six-jaw grapple designed to engage, lift, lower, and release MCOs and impact absorbers. Each of the six jaws is operated via two pairs of pinned linkage arms (bell cranks). Each set of linkage arms is pivot mounted on either side of a plate jaw carrier, and the six jaw carriers are welded to a top plate and a base guide plate to make up the main body fabrication. The jaws are mounted via pivot pins at one end of the linkage arms while the other end of the arms is attached to the jaw-operating weight by a roller and bearing arrangement. The operating weight moves vertically with respect to the jaw carriers. This vertical movement is translated through the linkage arms to operate the jaws in such a way that an upward motion of the weight causes the jaws to move downward and inward, closing the jaws. The operating weight is held raised by a single-acting, spring-return, operating cylinder mounted on the top plate. To close the jaws, the air pressure is removed from the cylinder, allowing the cylinder spring to retract the cylinder piston and raise the operating weight. The jaws are opened (Figure A2-38) before full engagement with the MCO. If the MCO is offset within the storage tube, the weight of the grapple is used to force the jaws partially closed as the
base guide plate guides the grapple assembly into alignment with the MCO lift feature. The locking plate is located beneath the base plate and is connected to the lifting cap by three operating rods that are mounted in a way that ensures that the locking plate will be raised before the main grapple body. A mechanical lock on the locking plate is designed to ensure that the jaws physically may not be opened if the lock is engaged and the grapple is supporting a load.

The MHM MCO shear and drop prevention interlock system channels consist of interlocks and the sensors, relays, and associated electrical power contactors. The sensors (limit switches, resolvers, photoelectric switches, proximity sensors, or load cells) associated with a given interlock measure or detect a parameter associated with the physical state of the MHM, such as the position of one of its mechanical components. The electrical power control contactor controls whether or not power is made available to the motor or drive associated with the interlock upon demand by the operator. The power contactor forms an open circuit when the interlock logic is not satisfied and a closed circuit whenever the interlock permissive logic is met. All of the MCO shear and drop prevention interlocks are dual-channel interlocks whose power contactors are in series so that power will not be supplied unless the interlock logic of both channels has been satisfied. The two interlock channels are electrically independent and physically separate to satisfy the single failure-proof criterion.

The interlock logic is described below for each safety-significant interlock.

P2 X channel

INHIBIT: Crane bridge travel motors operating and crane trolley travel motors operating and shield skirt jack raising

UNLESS: IF MHM is in MCO mode OR impact-absorber-exchange mode, the tube plug hoist is fully raised and the tube plug grapple jaws are fully open and the plug cavity is empty

AND

IF MHM is in tube-plug-exchange mode, and the tube plug hoist is fully raised and tube plug grapple jaws are fully open and the plug cavity is empty and the MCO grapple is in contact with a load and the MCO weight is above minimum grapple-plus-tube plug weight limit

OR

IF MHM is in tube-plug-exchange mode, and the tube plug hoist is fully raised and tube plug grapple jaws are fully closed and locked and the plug cavity is occupied and the MCO grapple is not in
contact with a load and the MCO weight is below the maximum grapple-only weight limit.

Y channel

INHIBIT: Crane bridge travel motors operating and crane trolley travel motors operating and shield skirt jack raising

UNLESS: IF MHM is in MCO mode OR impact-absorber-exchange mode, the tube plug hoist is fully raised and the tube plug grapple jaws are fully open and the plug cavity is empty

AND

IF MHM is in tube-plug-exchange mode, and the tube plug hoist is fully raised and tube plug grapple jaws are fully open and the plug cavity is empty and the MCO grapple is in contact with a load and the MCO grapple jaws are closed and locked and the MCO grapple load is not an MCO

OR

IF MHM is in tube-plug-exchange mode, and the tube plug hoist is fully raised and tube plug grapple jaws are fully closed and locked and the plug cavity is occupied and the MCO grapple is not in contact with a load and the MCO grapple jaws are not closed.

P6 INHIBIT: Turret rotation, turret locking pin disengagement, and base locking pin disengagement

UNLESS: IF in MCO mode or impact absorber exchange mode, the MCO grapple is fully raised

OR

IF in tube plug exchange mode, the MCO hoist is at the tube plug raise limit.

P21 INHIBIT: Shield skirt jack lowering, turret rotation, turret locking pin disengagement, base locking pin disengagement, tube plug hoist operation, and MCO hoist operation

UNLESS: The crane bridge clamps are fully applied and trolley restraint pins fully inserted.
Interlock P2 consists of the MCO-operating-mode-select switch, two tube-plug-hoist-fully-raised limit switches (LFPFRX and LFPFRY); two tube-plug-grapple-jaws-fully-open limit switches (LFPGOX and LFPGOY); two plug-cavity-empty photoelectric switches (PESSPX and PESSPY); two MCO-grapple-in-contact-with-a-load limit switches (LFMOCX and LFMOCY); two tube-plug-grapple-jaws-fully-closed limit switches (LFPGLX and LFPGLY); two tube-plug-grapple-jaws-locked limit switches (LFPGLX and LFPGLY); one MCO-grapple-jaws-not-closed limit switch (LFMJC); two MCO-hoist load cells (LCMHR1X and LCMHR2X); a trip-point unit connected to the output of the load cells; logic relays; and three sets of power contactors — one to the bridge travel motors, one to the crane trolley travel motors, and one to the shield skirt jack raise motors. The tube-plug-hoist-fully-raised limit switches are located inside and near the top of the tube plug hoist screw jack bellows assembly. The tube-plug-jaws-fully-open limit switches are positioned on a cam near the top of the actuator shaft assembly that rotates to open or close the tube plug jaws. The plug-cavity-empty photoelectric switches are located in the side wall of the tube plug cavity. The MCO-grapple-in-contact-with-a-load limit switches also are located on the MCO grapple assembly and are actuated when the grapple is seated on an MCO and the proximity probe is fully depressed. The tube-plug-grapple-jaws-fully-closed limit switches are positioned on a cam near the top of the actuator shaft assembly that rotates to open or close the tube plug jaws. The tube-plug-grapple-jaws-locked limit switches are positioned on a cam that rotates to open or close the tube plug jaws. The MCO-grapple-jaws-not-closed limit switch is located on the MCO grapple assembly and actuated by the position of the grapple central post. The two MCO hoist load cells are mounted on rope 1 and rope 2 of the MCO hoist. The sensors provide electrical input to the logic relays. If all required logic is satisfied, power is made available through pairs of power contactors, upon demand, to the bridge drive motors, the trolley drive motors, and the shield skirt jack motors for raising.

Interlock P6 (MCO hoist raise limit) is comprised of the following components: a limit switch (LFMGUDX) and associated push rod, a resolver (RSVMGHY and PLS2), logic relays, and three pairs of electrical power contactors. The P6 interlock limit switch is located in the MCO hoist compartment. A push rod that extends from the hoist compartment below into the MCO cavity activates the limit switch when the fully raised MCO grapple contacts the push rod. The resolver is mounted on the MCO hoist drive unit. The resolver constantly measures the height of the MCO grapple by measuring the angle through which the MCO hoist drive has rotated. The limit switch and the resolver provide separate electrical input to the two channels of logic relays. If required two-channel logic is satisfied, then power is made available, upon demand, through pairs of power contactors to the turret rotate motor, the turret locking pin motor, and the base locking pin motor.

Interlock P21 (bridge seismic clamps and trolley seismic restraints fully applied) is comprised of the following components: two sets of trolley dual limit switches (LFTSLP1AX and LFTSLP1AY, LFTSLP2AX and LFTSLP2AY), two sets of bridge limit switches (PSLTSC1AY and PSLTSC1AX, PSLTSC2AY and PSLTSC2AX), relays, and six pairs of electrical power contactors. The interlock P21 trolley seismic restraint pin limit switches are mounted on the trolley seismic restraint assembly. The bridge seismic clamp detection limit switches are mounted
inside the rail Johnson clamps. All of the limit switches provide separate electrical input to the
two-channel logic relays. If interlock P21 and all other required two-channel logic are satisfied,
then power is made available, upon demand, through pairs of power contactors to the shield skirt
jack motors for lowering, the turret rotation motor, the turret locking pin motor, the base locking
pin motor, the tube plug hoist motor, and the MCO hoist motor.

A4.4.4.3 Functional Requirements. The MHM structure and MCO hoist are designed to
prevent overstress or failure that could result in damage to the MCO or the CSB operating deck
during normal operation or during the DBE. The DBE seismic response spectra curves are shown
on Figure A4-4. The MHM safety-significant interlocks are designed to prevent an accidental
drop or shear of an MCO and provide radiation protection for workers in the CSB operating area.

A4.4.4.4 System Evaluation. The main structural components of the MHM will withstand the
DBE loading without the stress levels exceeding the allowable stresses permitted by
ASME NOG-1-1995. The turret and turntable assembly are designed to ASME NOG-1-1995,
Type I, crane requirements for permissible materials and stresses. The expected stresses for the
turret and turntable assembly have been demonstrated to be within code allowable values.

A dynamic analysis using the Response Spectrum Method was employed for the seismic analysis
of the MHM. This seismic analysis included the MHM grapple system. The response spectra for
accelerations acting on the MHM runway level were evaluated for a DBE having a zero period
acceleration 0.35 g horizontally and 0.23 g vertically above 33.33 Hz (see Figure A4-1). The
response spectrum method assumes the MHM responds as a linear elastic system. Results of the
seismic analysis are contained in report ESL/R(96)083.

The tube plug hoist and grapple have been designed to ASME NOG-1 Type-II crane
requirements for permissible materials and stresses. The calculated results of the stresses induced
in the load path components are contained in ESL/R(96)099. The MCO grapple is designed as a
special lifting device in accordance with ANSI N14.6-1993 for permissible materials, lower stress
allowables, and higher proof load requirements. This results in high safety factors for the rated
load. The ANSI N14.6-1993 safety margins require that analyzed stresses be below material yield
strength when multiplied by 6, or below material ultimate tensile strength when multiplied by 10.
See Table A4-6 for grapple load path components. The expected stresses for the MCO grapple
have been demonstrated to be within code allowable values. Proof testing of the grapple was
performed at 3 times the rated load of 12 tons (72,000 lb) in accordance with ANSI N14.6-1993.

The MCO hoist and grapple designs have been shown to be single-failure-proof against a
load drop event. The hoist is equipped with two brakes, redundant ropes and one drive train. The
interlocks discussed previously also terminate some events, which could lead to a load drop event
such as overspeed or hoist operation above upper limit. The MHM vendor performed failure
mode and effects analyses on both the MCO hoist and grapple and the MCO hoist and grapple
control system. The analyses are documented in ESL/R(97)36 and ESL/R(97)43, Failure Modes
and Effects and Hazard Analysis of Hanford MCO Hoist and Grapple Control System. These
failure mode and effects analyses were carried out to the format of MIL-STD-1629A and
considered failures at the component level. The failure mode and effects analyses revealed the
components whose failures give rise to a dropped MCO. The results indicate that with implementation of strict setting-up and maintenance programs, the design meets the single-failure criterion.

Evaluation of the MHM contained in the seismic analysis report [ESL/R(96)083] also has determined that during a DBE, MHM movement is within the design tolerance, the MCO will not become trapped, and the MHM will not hammer or collapse on the operating deck.

The MCO shear and drop prevention interlock system is comprised of a number of sensors (resolvers, limit switches, photoelectric switches, load cells, proximity sensors), relays, power contactors, and other electrical components. The devices are initially set during preoperational acceptance testing and are subject to surveillance checks at regular maintenance intervals as recommended by the MHM vendor.

Figure A4-5 is a single-line power distribution diagram that depicts the arrangement of the MHM power and control interlock system. Facility power is provided to the MHM at 480 V/3 ph. AC. The power passes through the seismic trip panel, through a main disconnect, and delivered through the trolley festoon cable to control cabinets located on the MHM trolley. The power is provided through fused 480/120 VAC transformers to separate power and interlock panels for the X and Y channels. The X-channel cabinets also include control circuitry from the operator’s console. Relay logic is powered by 120 VAC. When control power is lost to one or both channels, any single failure will not defeat both the X and Y channel. If all power is lost, none of the drives will operate.

The MHM interlocks protecting against an MCO drop are active interlocks in that they perform an action. They are designed to interrupt MCO hoist operation when unfavorable conditions are encountered or prevent MCO grapple operation. A generic interlock is shown in Figure A4-6. It consists of an X-channel and a Y-channel. Each channel consists of sensors, logic relays and a power contactor to the hoist drive motor. All components in the channel have an active safety function to perform when control and drive power are available. There is no safety function to perform when drive power is removed.

In general, two sets of redundant and independent sensors are used, one for each channel. The X and Y channel control power devices, relay logic and power contactors are located in separate panels dedicated to the X and Y channels as shown on Figure A4-5. These panels and the control console are mounted around the periphery of the trolley. A fire in one panel or failure in a panel cannot propagate to another panel. The panel supports are designed to ensure that the panels stay attached to the trolley frame and flooring during the DBE. There will be no missiles or falling components from the MHM that could sever the conduit and wiring between the panels or between the panels and the turret, trolley or bridge. The X-channel cabinets include part of the command circuitry for the drives coupled with the interlock logic. A command signal and satisfied interlock logic are necessary to satisfy the X-channel permissive. The command circuitry is powered by X-channel 120 VAC power and interfaces with the interlock logic through
energized relays. Closure failure of these relays has no effect on the remainder of the interlock logic and therefore the safety function is not compromised.

The MHM elementary control schematics will delineate the safety significant portions of the system. Periodic surveillance testing will ensure that the sensors, relays or power contactors have not failed closed. The operator panel is also equipped with status lights to show when the MCO hoist or MCO grapple are energized. There are no shared systems associated with the interlocks. In most cases, multiple interlocks will need to be satisfied before a power contactor to the MCO hoist drive motor or MCO grapple solenoid will be energized. The hoist interlocks are somewhat different than the passive permissive interlocks. In some cases, one channel will allow hoist motion in one direction for recovery operation when the second channel is manually overridden. A failure of the manual override only affects the one channel and has no impact on the other channel retaining the safety function.

A loss of electrical power or a seismic event exceeding the setpoint trips all control and drive power to the machine. The MHM is not required to perform any operation in the event of loss of electrical power or after the DBE. Regardless of MCO position, there is adequate natural convection cooling to prevent MCO temperature limits from being exceeded. Any of the drives can be manually actuated to rotate the turret, raise or lower the MCO, etc, should this become desirable. When power is restored, the machine power must be manually reset and a startup sequence performed before any drive can be actuated; therefore, the machine will not spontaneously resume hoist operation when power is restored. The behavior of the system for earthquakes less than one half the DBE is discussed in Section A2.5.1.3 under the seismic trip system evaluation.

The following paragraphs describe the various safety-significant interlocks associated with the MCO.

The sensors that are part of interlock P2 must be functionally evaluated to ensure that the MHM cannot remove a storage tube plug from a storage tube without replacing it with the same or another plug before raising the shield skirt to relocate.

- The pair of tube-plug-hoist-fully-raised limit switches are positioned near the top of the tube plug hoist screw jack bellows assembly and directly contact the top of the hoist when activated.

- The two tube-plug-grapple-jaws-fully-open limit switches are set to indicate that the jaws are closed only when the cam that opens the jaws is within 5° of the fully open rotational position.

- The pair of plug-cavity-empty photoelectric switches simply indicate whether or not a tube plug is in the tube plug cavity.
The pair of MCO-grapple-in-contact-with-a-load limit switches are set so that the grapple proximity probe must be essentially fully extended for no contact to be sensed. The X channel makes contact when the proximity probe has been depressed 4 mm or more, and the Y channel detects a load if the proximity probe has been depressed 3 mm or more from its fully extended position.

The fully closed jaw position will have a maximum separation between opposing jaw tips of 2.5 in.

The two MCO hoist load cells circuitry must be calibrated so that when the weight suspended from the MHM grapple is between zero and 24,000 lb, the reported weight indication has a minimum accuracy of 5% of rated load (HNF-S-0468).

All of the preceding checks on components ensure that they are calibrated or set and are not malfunctioning in a way that could bypass the intent of interlock P2.

Interlock P6 (MCO hoist at raise limit) requires that both a limit switch and the MCO hoist interlock resolver and programmable limit switch indicate that the MCO hoist is fully raised. The push rod that activates the limit switch has an adjustable beveled sleeve that contacts the limit switch. The switch will not be actuated unless the top of the MCO grapple assembly is less than 5.5 in. below the outside bottom of the MCO hoist and drive assembly box. The MCO hoist resolver is set such that it will not indicate that the MCO hoist is fully raised unless the grapple is less than 5.5 in. below the outside bottom of the MCO hoist and drive assembly box.

Interlock P21 (bridge seismic clamps and trolley seismic restraints fully applied) requires that two sets of dual limit switches on the trolley seismic restraint pin indicate that the restraint is fully applied. Interlock P21 also requires that four sets of dual pressure switches on the bridge seismic clamps indicate that the clamps are fully applied. These switches are adjusted so that they do not indicate that the clamps have been fully applied until both sides of the rail clamps are simultaneously in contact with the sides of the rail (under the railhead).

A4.4.4.5 Controls (Technical Safety Requirements). The following assumptions associated with the MHM require TSRs to ensure performance of the safety functions.

- MHM interlocks prevent shearing an MCO from translational movement. Interlocks P6 and P21 are operable.
- Configuration control of the MHM is maintained to ensure its continued structural integrity.
- To ensure that the MHM radiation protection interlock system is operable, operational checks of interlock P2 must be performed consistent with the frequency recommended by the MHM supplier.
A4.4.5 Tube Vent and Purge Cart

A4.4.5.1 Safety Function. The safety function of the pressure boundary portions of the tube vent and purge cart is to maintain confinement when connected to an overpack storage tube plug and minimize releases of particulate from overpack storage tubes to the environment. The tube vent and purge cart is provided for use in recovery operations involving MCOs should the recovery operations team deem its use appropriate. The tube vent and purge cart (see Figure A2-47) provides a means to periodically purge the overpack storage tubes of hydrogen and to sample the gas stream for concentrations of radioactivity. The cart contains a helium gas supply to install and maintain an inert atmosphere around the MCOs staged in the overpack tubes. The tube vent and purge cart also provides confinement when connected to the overpack storage tube plug (see Figure A2-14). Confinement is ensured by providing a HEPA-filtered pathway for controlled venting of gases that may have built up in the overpack storage tubes from relief of pressure from MCOs. To minimize the potential for release of particulate to the operating area, the cart contains HEPA filtration equipment to remove particulate released into the overpack storage tube and pushed into the cart.

Other non-safety related functions of the tube vent and purge carts associated with preparing the standard and/or overpack storage tube for removal or placement of an MCO are removal of the tube plug cover assemblies and placement of the interface guide ring funnel to comply with TSRs in Sections A4.3.15 and A4.3.16. These operations are considered non-safety related and are further described in Section A2.5.1.3.

A4.4.5.2 System Description. The tube vent and purge cart is designed to monitor the gas concentration in the overpack storage tube, pressure purge the tube with helium while maintaining confinement (gases are vented through HEPA filters), and refill the tube with helium. The tube vent and purge cart is shown in Figure A2-47. A towable hoist is provided to remove the center access plate in the overpack tube plug. Figure A2-48 represents a diagram of piping, instruments, and equipment required by the tube vent and purge cart to service the storage tubes.

The tube vent and purge cart connects to the overpack storage tube plug by means of a 1.5-in., flexible metal hose. The pressure in the overpack storage tube, or the pressure introduced by the cart's onboard helium supply, pushes the tube atmosphere through the HEPA filter for removal of particulate exhausted from the storage tube. The metallic filter medium is rated for 175 °F, a flow rate of 20 standard ft³/min at a pressure drop of 1 in. w.g. The gas in the overpack storage tube is monitored by means of a 0.75-in. line from the 1.5-in. piping header through a 21 W heat exchanger, an in-line continuous air monitor, radioactive-gas monitor, and hydrogen monitor. Instruments on the tube vent and purge cart detect airborne radioactivity and hydrogen concentrations in an overpack storage tube.

A 0.75-in. manifold feeds inert gas to the 1.5-in. cart header at 5 lb/in² gauge from two 2,000 lb/in² cart-mounted gas bottles. The inert atmosphere is initially supplied from these high-pressure inert gas bottles after the overpack storage tube has been vented. A pressure regulator and relief valve prevent overpressurization of the overpack storage tube.
The tube vent and purge cart is battery powered for motion. The equipment on the cart is powered through cables inserted into electrical outlets in the deck. The tube vent and purge cart equipment is operated using manually operated valves that are located on the cart. The operator has to make and break connections to the overpack storage tube plug. The overpack storage tube plug is designed to provide shielding, confine radioactive particulate, and minimize hydrogen fire hazards (see Sections A4.4.3 and A4.3.5).

The safety-significant components of the tube vent and purge cart include all lines, fittings, valves, components, and instruments between the overpack storage tube plug connection and the outlet from the tube vent and purge cart HEPA filter. These items ensure that the overpack storage tube atmosphere remains inert by excluding air ingress during all tube vent and purge cart operations.

**A4.4.5.3 Functional Requirements.** The tube vent and purge cart is used to install, maintain, and replenish the inert gas atmosphere in the overpack storage tubes. The cart facilitates controlled venting of gases that may have built up in the overpack storage tubes from relief of pressure from the accident or suspect MCOs (these MCOs may be leaking through the gasketed cover plates bolted to the four ports of the cover assembly). Gases from the overpack storage tube are monitored and vented to the operating area through a HEPA filter on the tube vent and purge cart. The cart interfaces with a connection on each overpack storage tube plug that allows monitoring and purging of the tube atmosphere without interfering with the sealing surfaces of the plug. The cart’s flexible hose and cart piping and equipment must be designed to withstand the expected maximum overpack storage tube atmosphere temperature and pressure. Overpack storage tube design pressure is 75 lb/in² gauge (see Section A4.3.5). Figure A4-3 indicates a tube exterior temperature of 186 °F for the standard storage tube containing two MCOs. Overpack storage tubes are designed to contain only one MCO. Gas temperatures are expected to be lower for the overpack storage tubes.

The tube vent and purge cart vents, purges, and refills an overpack storage tube to maintain a slightly positive pressure. Equipment on the tube vent and purge cart can detect airborne radioactivity and hydrogen concentrations and provide HEPA filtration of the sampled volume before release to the operating area atmosphere. The cart equipment takes a sample of the overpack tube’s atmosphere (the gas temperature could range from 125 °F to 186 °F as shown in Figure A4-3), passes it through the monitors, and vents it to the operating area through two stages of HEPA filters. Connections to the overpack storage tube are made using quick-disconnect fittings atop the storage tube plug.

**A4.4.5.4 System Evaluation.** Piping and the external surfaces of other equipment (e.g., gauges, valves, HEPA filters) are adequate to provide confinement of gases from the overpack storage tubes under any potential venting conditions. Components not required to ensure confinement or to minimize the potential for ingress of air during tube servicing are considered general service features.
The tube vent and purge cart will fulfill its safety function of maintaining safe conditions in overpack storage tubes in the following way. After initial inerting, the pressure in an overpack storage tube is slightly positive (4 lb/in² gauge). If hydrogen is relieved from the MCO, the pressure in the overpack storage tube increases. The hydrogen concentration and pressure are measured each time the tube vent and purge cart makes a scheduled visit to an overpack storage tube. If an MCO relief has occurred, the hydrogen concentration could exceed 4%; in this event, the hydrogen concentration in the overpack storage tube will be diluted to below 4% by adding inert gas before the release is vented to the operating area. This process also accomplishes the safety function of preventing oxygen ingress into the overpack storage tubes.

The pressure rating of the tube vent and purge cart components of 150 lb/in² gauge and 400 °F as listed in Table A2-6 are adequate to ensure that the overpack storage tube inert gas atmosphere operation does not exceed the purge cart temperature ratings and is accomplished in a safe manner.

A4.4.5.5 Controls (Technical Safety Requirements). There are no TSR controls for this SSC because its safety function is related to recovery actions. Controls related to specific recovery actions must be developed on a case-by-case basis depending on current conditions and recovery evaluation results.

A4.4.6 Receiving Crane Structure and Hoist

A4.4.6.1 Safety Function. The safety function of the receiving crane structure and hoist is to provide safe handling of the transportation cask and MCOs. The receiving crane and hoist are designed to safely handle transport casks loaded with MCOs. The receiving crane and hoist are designated safety significant in accordance with Letter 97-SFD-172 (Sellers 1997) and DOE-STD-3009-94 and ITS Category B per NRC equivalency requirements using the graded approach in HNF-SD-SNF-DB-003. The safety functions for which the receiving crane is credited are described in Section A3.4.2.1.

After removing an MCO from the transportation cask (as described in Section A2.5.1.3 and shown in Figure A2-32) an empty MCO shell (without baskets or shield plug) is placed into the empty transportation cask, the cask lid is put back on and the completed assembly repositioned onto the transport trailer. The receiving crane is used for placing the cask containing an empty MCO onto the transport trailer. This operation is analyzed in HNF-SD-SNF-HIE-001, Canister Storage Building Hazard Analysis Report, Table 19, checklist designator SA-G-03a.

A4.4.6.2 System Description. The receiving crane is a 60-ton capacity, equal-leg gantry crane with two wire-rope hoists (a main and an auxiliary) supported on a top-running trolley. The receiving crane is designed in accordance with ASME NOG-1-1995 Type I, and Crane Manufacturer’s Association of America class D for indoor service. The receiving crane gantry spans 27 ft, 6 in., and it can travel 164 ft east–west. The receiving crane includes a 10-ton auxiliary hoist, a shielded operator station, remote operating equipment including a powered
rotating hook and local control. Both the bridge and trolley speeds are variable up to 40 ft/min.

Hoist speed for both the main and auxiliary hoists is variable up to 10 ft/min. The trolley weight is 43,000 lb, and the bridge weight is 156,000 lb. Maximum wheel load is 135,000 lb. The NEMA 12 control enclosure is designed in accordance with NEMA 250, Enclosures for Electrical Equipment (1000 Volt Maximum). The power supply is 460 VAC, 3 ph, 60 Hz.

The receiving crane is used to off-load transportation casks from their vertical position on the transport vehicle and to place them into the cask receiving pit. After the MCO has been removed from the transportation cask, the receiving crane is used to place a new (empty) MCO into the transportation cask, remove the transportation cask from the cask receiving pit, and return it to the transport trailer. The transport trailer is loaded and unloaded while it is located in the trailer vestibule. The trailer vestibule is an extension of the CSB that provides an air-locked transition area for the transport trailer and receiving crane to off-load the transportation cask (see Figures A2-19 through A2-21, and A2-31).

The receiving crane is equipped with a shielded operating station located on the outboard side of the north gantry leg. This station is necessary to implement ALARA considerations. Crane control is available either from the operating station or by radio. The receiving crane is expected to receive a total integrated dose of $4.0 \times 10^3$ rad over its 40-year design life. The shielded operating station is 3.5-in.-thick carbon steel with viewing windows at least 12 in. by 6 in. in size providing a normal viewing angle of 50° in the horizontal and vertical direction and an extreme viewing angle of 90° in the horizontal and 80° in the vertical direction.

A4.4.6.3 Functional Requirements. The receiving crane operates over the cask receiving pit, which may contain an MCO and a containment tent. The receiving crane features that prevent dropping loads or components onto the MCO are designated safety significant. DOE Order 6430.1A, Section 1320-5, requires systems and equipment that “ensure safe loading, removal, and handling” of SNF to be designed as “safety class items.” The receiving crane does not handle unconfined SNF nor carry loads over unconfined SNF. Therefore, the features of the receiving crane that prevent crane collapse, dropped loads, or falling components from breaching the MCO are designed to safety-significant criteria (Williams 1998b). Features of the receiving crane that provide protection against tipping over and potentially impacting the safety-class operating area deck structures are designated safety class and required to be designed to withstand the DBE with factors of safety per ASME NOG-1-1995.

Not all receiving crane components need to be operational after a seismic event. If a suspended transportation cask remained inaccessible for sufficient time, overheating of and/or hydrogen buildup in the contained MCO could result in a release of radionuclides to the interior of the transportation cask. The transportation cask is designed to prevent damage to the MCO and to prevent the release of radioactive material if a drop were to occur. Analyses referred to in SNF-3328, Canister Storage Building Design Basis Accident Analysis Documentation, address the consequences of a dropped transportation cask in the CSB.
Receiving crane load rating of 60 tons (120,000 lb) is based on the expected future mission of servicing FFTF fuel casks (see HNF-S-0425). The required rating for the SNF mission is approximately half of this value. The transportation cask containing a full MCO weighs less than 60,000 lb (see HNF-SD-TP-SARP-017). The shielded operator viewing window is required to reduce 1 MeV gamma radiation by a factor of 50.

**A4.4.6.4 System Evaluation.** During normal operations or a DBA, receiving crane components are designed to prevent the drop of a transportation cask containing an MCO, thus preventing severe damage to the transport vehicle, the reinforced concrete floor, and/or the cask receiving pit structure.

The receiving crane is designed to resist tipping over and provide protection against a DBE. The receiving crane is considered decoupled from the runway rails (e.g., no mechanical rail clamps). Lateral displacement is restrained by the wheel flanges and longitudinal displacement is restrained by the crane wheel brakes during a seismic occurrence. The receiving crane is stable against overturning during a seismic occurrence with a safety factor of 1.1, in accordance with ASME NOG-1-1995. Results of the stability analysis performed by the manufacturer of the receiving crane are provided in WRC1-AC-0014, *Seismic Analysis and Design of the 60/10 Ton Receiving Crane with the Credible Critical Load*. The results indicate that the minimum safety factor against overturning for the extreme environmental load conditions is 1.64 about the x axis (transverse horizontal) and 1.1 about the z axis (transverse longitudinal). Static and dynamic analyses of the receiving crane have been performed to verify that the receiving crane will not tip over or collapse during a seismic event (WRC1-AC-0014). The seismic design ensures that the crane will not tip over and impact the safety-class MCO located in the cask receiving pit, or other safety-class or safety-significant SSCs, or cause injuries or fatalities to facility workers.

The receiving crane was also evaluated for the extreme environmental load combination considering dead, live, and seismic loads. The load path components (Figure A2-19) used in the stress analysis for the receiving crane include the trolley, bridge, legs, end trucks, hoist drums, girders, and beams. The live load is the credible critical load of 60 tons. The seismic analysis of the crane is based upon the provisions of ASME NOG-1-1995, Section NOG-4150. A linear response spectrum analysis is used for the evaluation, which is documented in WRC1-AC-0027, *Seismic Analysis and Evaluation of the 60/10 Ton Receiving Crane with the Credible Critical Load*. The static and dynamic analyses of the receiving crane are performed with the STARDYNE computer program. Four STARDYNE models represent the receiving crane and trolley in order to satisfy the requirements of ASME NOG-1-1995, Section NOG-4153.7. At each trolley position, an analysis is performed considering the credible critical load in the raised and the lowered position. The seismic input is based upon the response spectra included in Figure A4-7. The modal responses are combined based upon the Ten Percent Method permitted by ASME NOG-1-1995, Section NOG-4153.10(b)(2). The percentage of critical damping is not required by this modal combination method. The three-directional components of the structural response to earthquake motion are combined by taking the square root of the sum of the squares of the responses caused by each component of earthquake motion as described in ASME NOG-1-1995, Section NOG-4153.10(c). As indicated in the seismic analysis report (see
WRC1-AC-0027), the demand capacity factors for combined bending and axial load for the
girders, endties, legs, bridge endtrucks, and trolley endtrucks are less than 1.0. The maximum
demand-to-capacity ratio for the trolley rail is 0.81 based upon the requirements of ASME
NOG-1-1995, Section NOG-4423.4. The demand-to-capacity ratio for the trolley main load girt
is 0.66 based upon the requirements of ASME NOG-1-1995, Sections NOG-4312, NOG-4313,
NOG-4321, and NOG-4322. The design of bolted connections is based upon the Research
Council on Riveted and Bolted Structural Joints of the Engineering Foundation’s “Structural
Joints Using ASTM A 325 or A 490 Bolts.” The allowable stresses for bearing type joints are
increased by a factor of 1.50 for extreme environmental (DBE) load conditions. The allowable
stresses in welds are based upon AWS D1.1, Structural Welding Code — Steel, for operating load
conditions. The allowable weld stress may be increased by a factor of 1.5 for extreme
environmental conditions. The bolts associated with the connections at the top of the legs are
stressed to capacity for the extreme environmental load conditions. The maximum demand-to-
capacity ratio for these bolts is 1.029. This is acceptable because the compression is
conservatively included in the bolt force. The 0.625-in. welds located between the leg and the
connection plate and the connection plate thickness are adequate for the extreme environmental
load conditions. The bolts associated with the connection of the legs to the bridge endtruck have
a demand-to-capacity ratio of 0.970 (see WRC1-AC-0027).

The receiving crane hoist brake and the seismic accelerometer power-disconnect switch
provide safe shutdown of the receiving crane during a postulated seismic event. This provision is
a requirement of ASME NOG-1-1995 and not driven by safety analysis. If the accelerometer
detects a seismic event in excess of the seismic trip point, the power-disconnect switch
automatically causes loss of power to the receiving crane. On loss of power, the receiving crane
stops, and the load brake is applied preventing motion of either the receiving crane or a suspended
cask. Any suspended load remains in a safe position until power is restored and the operator
restarts load movement. These safety features prevent dropping of the cask if it is suspended
during a seismic event.

HNF-SD-SNF-FHA-002, Final Fire Hazard Analysis for the Canister Storage Building,
determined that a fire, in the load-in/load-out area or elsewhere, will not result in the loss of safe
handling of the cask-MCO by the receiving crane. Collision prevention switches are designed to
prevent the possible collision of the receiving crane with the MHM crane, as these cranes travel
on intersecting rails.

Bridge stops and over-travel limit switches prevent the receiving crane trolleys from
impacting the ends of the rails. Over-travel limit switches cause the trolley motors to slow down
and stop before impact with the bridge stops. The bridge stops are designed to absorb whatever
momentum is left after the over-travel limit switches have engaged. Collision prevention switches
prevent a possible collision between the crane and the trailer vestibule inner door.

The receiving crane system is designed, fabricated, manufactured, inspected, and tested in
accordance with the requirements of CMAA 70-94; ASME B30.20; and ASME NOG-1-1995,
Type I, with the exception of Section NOG-5420(a) (receiving is in accordance with

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CMAA 70-94) and Section NOG-5428.1 (a single hook is used rather than a double hook). The crane's main and auxiliary hoists were tested to 125% of rated load for the entire operating range, as documented in W379-012, Cask Receiving Crane PAT. An analysis and review of the cask and trunnion in HNF-SD-SNF-FDR-003, Design Analysis for the NT and WHC Cask and Transportation System, and an analysis of the cask lifting yoke in HNF-SD-SNF-DA-022, Report on Canister Storage Building (CSB) Crane Hook Structural Analysis, demonstrate that the cask lifting equipment complies with ANSI N14.6-1993 requirements.

The receiving crane design requirements also satisfy the additional NRC equivalency requirements of ANSI/ANS-57.1-1992 and ANSI/ANS-57.2-1992. Design features meeting the additional ANSI/ANS-57.1-1992 requirements include provisions for control manipulation with gloved hands, an interlock that prevents simultaneous hoisting and translational movement (except when manually actuated), and a conveniently located manual disconnect for emergency shutdown of the electrical power supply. Design features meeting the additional ANSI/ANS-57.2-1992 requirements include provisions for safe access to all components for the purposes of testing, inspection, and maintenance; a positive braking or holding mechanism (stop blocks) for the trailer vehicle in the facility; and limiting the potential height of a cask drop by electrical or mechanical controls. Results contained in Calculation CALC-ME-007, Shielding, demonstrate the adequacy of the shielded operator's console and viewing window.

A4.4.6.5 Controls (Technical Safety Requirements). There are no TSR controls for the safety-significant receiving crane structure and hoist for this safety function because the credited features are design features (see Section A5.6) and their safety function is managed by the configuration control process.

A4.4.7 Cask Receiving Pit

A4.4.7.1 Safety Function. The safety function of the cask receiving pit is to protect the facility worker from excessive doses of radiation during the time that the MCO transportation cask is in the cask receiving pit with the cask lid removed and while the MCO is being removed from the transportation cask by the MHM.

A4.4.7.2 System Description. The cask receiving pit is a reinforced concrete structure integral with the 5-ft-thick load-in/load-out concrete floor described in Section A2.4.2. The concrete floor is supported at grade. The cask receiving pit is a 56-in.-diameter, 18.5-ft-deep cylinder. The top 4 ft of the pit contain a 1-in.-thick carbon steel liner embed assembly welded to concrete embedded weld studs. The liner embed assembly carbon steel plate conforms to ASTM A36/A36M. Cask guides mounted in the cask receiving pit center the transportation cask as it is lowered into the pit.

A4.4.7.3 Functional Requirements. The cask receiving pit is designated safety significant. This pit and the shield hatch and MCO assembly prevent radiation exposure by sealing to the MHM while an MCO is being removed from the transportation cask and lifted into the MHM.
The shielding goal for the cask receiving pit and the shield hatch and MCO guide assembly is to meet the ALARA guidelines of Title 10, *Code of Federal Regulations*, Part 835, “Occupational Radiation Protection (10 CFR 835). The cask receiving pit is designed to meet the requirements for personnel shielding safety in accordance with HNF-SD-SNF-DB-003 (Items 20 and 23).

### A4.4.7.4 System Evaluation

The cask receiving pit and the safety-significant shield hatch and MCO guide assembly provide a radiologically safe area for operators to conduct cask receiving and MCO retrieval operations. With the shield hatch and MCO guide assembly in place, radiation dose rate to the facility operator is less than the 0.05 mrem/h calculated value for the operating deck over vault 1 (CSB-SH-2002). The pit is adequate to protect the facility worker and complies with Items 20 and 23 of HNF-SD-SNF-DB-003.

### A4.4.7.5 Controls (Technical Safety Requirements)

The cask receiving pit is a design feature (see Section A5.6), and its safety function is managed by the configuration control process. Design features of the CSB structures ensure that their safety function can be relied upon.

### A4.4.8 Transportation Cask Servicing System

#### A4.4.8.1 Safety Function

The safety functions for the transportation cask servicing system are to protect an MCO from exceeding external design pressures, maintain confinement when connected to the transportation cask, minimize releases of particulate from the transportation cask, and prevent buildup of flammable gases in the transportation cask. The transportation cask servicing system prevents or mitigates a radiological release by purging generated gas (primarily hydrogen generated from the MCO) from the cask and replacing the purged gas with helium. Subsequently, if required, the suspect MCO can be removed from the cask and placed in an overpack storage tube for monitoring by the MCO tube vent and purge cart. Servicing of normal shipping casks and MCOs arriving from the Cold Vacuum Drying Facility is not required (i.e., whenever the internal cask pressure is less than 3.0 lb/in² gauge). Refer to HNF-4509, *Evaluation of Canister Storage Building Cask Receipt Pressure*, for additional details.

The majority of the items in the transportation cask servicing system are classified as safety significant because of their role in providing confinement and worker safety. The transportation cask servicing system is designated safety significant in accordance with DOE-STD-3009-94 and implements the guidance of HNF-PRO-704 for design codes and standards. This system has not been identified in Chapter A3.0 as mitigating or preventing an accident or DBA. It is provided for use during recovery from an accident or suspect MCO shipping cask should the facility recovery team deem it appropriate.

Overpressurization protection is provided by pressure safety valve PSV-102 located outside the west wall of the support building (see Figure A2-24). This valve is set to relieve at 150 lb/in² gauge.
A4.4.8.2 System Description. The transportation cask servicing system comprises an inert gas supply, flexible steel hose, a canister HEPA filter (FH-2), piping, valving, and control system interfaces. The transportation cask servicing system is designed to vent the cask to a safe condition by removing excessive amounts of hydrogen and replacing a high-hydrogen cask atmosphere with inert gas. This replacement is accomplished by repeatedly venting and pressure purging any cask that has been determined to be abnormally pressurized (>3 lb/in² gauge). The cask is vented and purged while located inside the cask receiving pit. The piping and instrumentation are depicted on Figure A2-24. The containment tent is available for control of radioactive contamination during recovery operations, if needed.

The designed flow path for servicing system operations begins with aligning the system for venting (confinement of gases vented from a transportation cask containing an abnormal MCO is controlled to safety-significant criteria). The quick-disconnect fitting is connected to the appropriate fitting on the transportation cask. The gas from the transportation cask is vented through a HEPA filter. Equipment provided for the venting operation downstream of the HEPA filter includes piping, valves and fittings, a sampling connection with a flow orifice, and an inert gas (helium) dilution control system. The sampling connection provides a means of pulling a sample from the transportation cask and is designed for the potentially high temperature of the gas coming from the transportation cask. The addition of inert gas to the transportation cask off-gas stream is controlled via a pressure controller combined with the flow orifice to promote mixing and dilution before venting the potentially high-hydrogen gases into the operating area and through the exhaust stack.

The designed flow path for the cask servicing system ends with aligning the system for the purging operation. Purging is performed by providing pressurized inert gas directly to the transportation cask. A combination of special-connection hardware specific to the inert gas and operation of the transportation cask servicing system at atmospheric pressure and above minimizes the potential for inadvertently adding air to the transportation cask. Upon completion of venting and purging operations, the transportation cask is returned to atmospheric pressure. Hydrogen monitoring can be performed using the tube vent and purge cart if necessary. The quick-disconnect fitting is disconnected from the appropriate fitting on the transportation cask before the cask lid is removed and the MCO removed from the service station by the MHM for transport to the sampling/weld area or a standard storage tube.

A4.4.8.3 Functional Requirements. Transportation cask purging and inerting operations must not create a flammable atmosphere inside the transportation cask. Using a closed (air-free) system to purge the transportation cask of hydrogen, and to return the transportation cask to atmospheric pressure, prevents the creation of a flammable atmosphere.

HNF-SD-SNF-DB-003, Item 12, requires that the FH-2 HEPA filter be in accordance with ASME N509-1989 and ASME N510-1989 to implement the principles of ALARA.

A4.4.8.4 System Evaluation. The transportation cask servicing system is designed as a closed system that operates at atmospheric pressure and above. Special connections unique to
pressurized helium gas bottles prevent inadvertent attachment of the wrong gas (oxygen or air) thus minimizing the potential for hydrogen deflagration resulting from use of the cask servicing system. Since the transportation cask is initially un-inerted, the consequences of accidentally introducing air into the cask are minimal. All system components that connect to the abnormal or accident transportation cask and that filter released gases are designed to 150 lb/in² gauge and 400 °F. The cask servicing system components that provide filtration and dilution of cask atmosphere are located in the support area building. See Table A2-2 for design and operating conditions of the cask servicing system.

During venting operations, the transportation cask is depressurized to atmospheric pressure. The offgas is vented to the HVAC exhaust stack through a filtered pathway, first passing through the sintered metal canister-type HEPA filter (FH-2) and then through a secondary nonmetal HEPA filter, as discussed in Calculation CSB-PR-0012, Tube Plug HEPA Filter Particulate Loading, eventually joining the HVAC exhaust airflow just upstream of the HVAC exhaust fans (see Figure A2-51). The FH-2 HEPA filter is 99.97% efficient and allows only 0.3 μm or smaller particle penetration with a design pressure range of full vacuum to 150 lb/in² gauge and a maximum design temperature of 400 °F. The FH-2 HEPA filter is in accordance with ASME N509-1989 and ASME N510-1989. The normal operating range of the HEPA filters is 0.1 lb/in³ absolute to 80 lb/in³ gauge with a maximum temperature of 150 °F. A 1.5-in. temperature element with a design temperature range of 50 °F to 400 °F and a maximum design pressure of 150 lb/in² gauge is used in the flow path to monitor and record the transportation cask gas temperature as the cask depressurizes. The design pressure for the vent piping is in accordance with ANSI/ASME B31.1, Power Piping, for the safety significant portion and ANSI/ASME B31.3, Process Piping, for the general service portion. The design temperature range is -27 °F to 400 °F. According to Calculation CSB-P-0001, Pipe Stress Calc for LN-MOC-038, 041, 042, the transportation cask servicing system components are designed to seismic ground acceleration of 0.21 g, which is appropriate for safety-significant items.

The inert gas system also plays a role in diluting the transportation cask exhaust gas to minimize the potential for a flammable gas mixture to occur in the 3-in. piping upstream of the secondary HEPA filter. This is accomplished by using a flow orifice and an inert gas pressure control system to ensure enough inert gas is mixed with the gas venting from the transportation cask to minimize the potential for a flammable concentration of hydrogen to enter the operating area exhaust system and exhaust stack. This system also is used to assist the vent system in providing an atmospheric pressure vent path (0.0 lb/in² gauge) and inert gas purge (80 lb/in² gauge). The venting and purging operations are repeated until the residual atmosphere in the transportation cask is below flammability limits. The helium system supplies the inert gas required to purge the transportation cask to safe hydrogen concentration levels and to mix and dilute gases as they are vented. Servicing an abnormal or accident transportation cask containing the maximum credible amount of hydrogen may require up to three vent-and-purge cycles to ensure that the residual concentration of hydrogen in the serviced transportation cask is not hazardous. This value is conservative considering the smaller freeboard volume of the transportation cask as compared to the MCO. Calculation CSB-PR-0013, Number of Pressure Purges Required, estimated the number of purges for an MCO at three. After purging operations are complete, the
transportation cask is returned to atmospheric pressure and monitored for acceptability before the
cask lid is removed and the MCO is removed with the MHM.

A4.4.8.5 Controls (Technical Safety Requirements). There are no TSR controls for this SSC
because its safety function is related to recovery actions, not to accidents during normal
operations. Controls related to specific recovery actions must be developed on a case-by-case
basis depending on current conditions and recovery evaluation results. The HEPA filter testing
and system component surveillances are performed by operations.

A4.4.9 Multi-Canister Overpack Sampling System

A4.4.9.1 Safety Function. The safety function of the MCO sampling system is to maintain
confinement when connected to an MCO, prevent a flammable mixture in an MCO or in the
sampling equipment, protect workers from radiation dose, protect the MCO from NPH and load
drop events. A bumper protects the sampling system from damage caused by equipment moving
horizontally. The MCO sampling system provides data for an engineering evaluation of SNF
stored in MCOs. This is accomplished by a program of sampling the gas inside MCOs selected
for monitoring and accumulating information from that sampling program. The safety-significant
features of the MCO sampling system are the sampling/weld station MCO support structure
(which includes the rotating shield, the stationary shielding, the shear ring and shear screws, and
the stationary shield support pipes), the sample hood and HEPA filter, the exhaust system from
the sample hood to and including exhaust fan HEPA filter, the sample hood flow indicator, shield
halves, center shield plate, and MCO valve operator. These features are designed to maintain the
confinement barrier for gases from an MCO during sampling operations to prevent the buildup of
flammable gases in the MCO or in the sampling/weld station equipment. Another safety function
related to the MCO sampling system includes shielding (shield halves, center shield plate, rotating
shield, and stationary shielding) to protect personnel from radiation exposure.

A4.4.9.2 System Description. The MCO sampling system, located in pit 7 in the sampling/weld
area of the CSB, is designed to sample and monitor selected MCOs. See Figure A2-3 for a
location in plan view. Duplicate support equipment at pit 2 is also available for sampling. The
major components of the MCO sampling system are the impact absorber used to cushion the
MCO if it is dropped by the MHM as it is being lowered into a sampling/weld station, a gantry
crane system for positioning the sample hood and temporary shielding components, shield halves
to guide a dropped MCO into the sampling/weld pit and protect operating personnel against
radiation streaming during MCO transfer, fixed shielding with MCO cooling capability, a cooling
cap to cool any high-temperature MCO shield plug assembly, a sample hood with a glove box to
confine airborne radiological emissions accidentally released during sampling operations, a
filter–exhauster to ventilate the hood, an infrared pyrometer to measure MCO skin temperature
when the MCO is lowered into the pit, an MCO valve operator for the process valve in the MCO
that provides gas sample access, an MCO sample cart for sampling and reinserting the MCO, a
sample syringe to extract a gas sample for analysis, tools for MCO process port removal and
cover plate removal and installation operations, and leak-test equipment to test the MCO cover plate seals.

MCOs are received at the sampling/weld station from either their preselected storage tubes or directly from the CSB load-in/load-out area. The MCOs are transported to the sampling/weld station by the MHM. The MCO sampling/weld station cart is used during all MCO sampling operations and is designed to provide a mobile storage location for the components required to (1) determine MCO internal pressure and gas temperature, (2) obtain a sample of the MCO internal atmosphere for laboratory analysis, and (3) refill the MCO with inert gas, if needed, following the sampling operations to ensure a positive pressure (nominally at 7 lb/in² gauge) during MCO storage.

The sampling process begins when the MHM brings an MCO to the sampling/weld station and lowers the MCO into the sampling/weld station pit until the MCO is resting on the safety-significant impact absorber with the MCO collar slightly above the shear ring on the rotating shield (see Figure A2-40). An infrared pyrometer scans the MCO while the MCO is being lowered into the pit, and the temperature readings are recorded and tracked by the distributed control system. A 7.5-ton gantry crane is used to remove the shield halves and center shield plate; the sampling/weld station trench cover shield is removed; and temporary handrails are placed around the trench. An operator enters the trench and uses a hand-held pyrometer to measure the MCO shield plug temperature. Based on these temperature readings, a cooling cap may be placed on the MCO shield plug to reduce the temperature to a value safe for operations.

The sampling hood is lowered over the MCO, and connections are made to the vent system and the sample cart. The sample cart’s flexible lines are connected to the electrical outlet and the helium supply line. Using the sampling procedure, the lines are purged, a sample is taken, and the MCO is reinerted with helium if needed. After sampling, the operator checks for leaks, and purges the sample cart lines. The vent system is shut down, the cart is disconnected, the vent system is disconnected, and the sample hood is lifted from the MCO. The removable guard rails are removed, the trench cover shield is reinstalled, the shield halves and center shield plate are reinstalled, and the MHM retrieves the MCO and returns it to the preselected storage tube or to a standard storage tube. This procedure for MCO sampling is designed for collecting data.

### A4.4.9.3 Functional Requirements

The MCO sampling system is required to prevent ingress of air to the MCO during sampling operations to prevent the formation of flammable concentrations of hydrogen outside the MCO around the process port connections under the sample hood. The lower shield support columns are required to maintain their structural integrity, including during a seismic event. The sampling/weld station impact absorber (IMP-005) is designated safety significant for its role in protecting the MCO and is described in Section A4.4.20.

The sample cart can be used to refill the MCO with helium to ensure a positive pressure during storage. The helium backfill pressure must be controlled to acceptable limits so as to not damage sampling/weld station components or cause an unmitigated release of particulate from the
MCO. The sampling equipment must be able to perform its safety function during normal and postulated NPHs and DBA conditions. Any features required to maintain the confinement barrier for the MCO during the sampling and pressure checking are safety significant.

The MCO sampling/weld station gantry crane is classified as general service (HNF-PRO-704) and ITS Category C in accordance with the definitions in HNF-SD-SNF-DB-003 because of the potential damage to the MCO process port covers that could be caused by dropping a shield half onto them. Once the MCO is placed in the sampling/weld station pit and the center shield plate is replaced, the MCO is protected from damage from falling objects. The MCO sampling/weld station gantry crane crash shield guard protects the MCO sample lines from a shear by the MHM by activating the MHM's collision avoidance system. The seismic input is based on the response spectra included in Figure A4-8.

The MCO sample hood is classified as safety significant because the hood is designed to maintain a confinement barrier around the top of the MCO during sampling operations and under certain design basis events that could allow detonation of hydrogen from a broken sample line. Other safety-significant designations are assigned to equipment credited in protecting the facility worker from serious injury or death as a result of radiation exposure. These components are the stationary and rotating shielding, the trench cover shields, the drive assembly cover shields, and the shield halves. The inverted cone shape of the shield halves opening guides an accidentally dropped MCO into the sampling/weld station pit and mitigates the effects of the fall.

A4.4.9.4 System Evaluation. The sampling/weld station hood is designed to protect operating personnel from gases that might escape from the MCO process port connection during sampling operations and form a flammable mixture with air. The exhaust system is designed to provide a minimum flow of 100 ft³/min through the sampling/weld station hood. The sampling/weld station hood inlet slots provide a maximum flow area of 0.35 ft². This results in a flow velocity of 286 ft/min into the hood. According to Calculation CSB-HV-0013, Sample Station and Future Weld Station Exhaust Ventilation System Pressure Drop Calc, with one glove missing out of the port, the flow velocity would be 182 ft/min, which is greater than the 100 ft/min required by ERDA 76-21, Nuclear Air Cleaning Handbook, for glovebox design.

The rupture disk associated with the helium supply system protects the MCO from pressurization to greater than 150 lb/in² gauge and is classified as safety significant. The MCO sampling system piping is of at least 150 lb/in² gauge design rating and meets ANSI/ASME B31.1 requirements for safety-significant portions, and ANSI/ASME B31.3 piping code requirements for general service portions.

The shield halves, the center shield plate, and the shear ring are used to reduce radiation streaming coming from the MCO. These components maintain radiation exposures within facility limits and ALARA principles during lowering and raising of the MCO within the sampling/weld station pit and while performing sampling operations. Shielding analyses documented in CSB-SH-3005, CSB Sample Station Shielding Calculation, and ALARA analyses documented in ALARA 09, SNF Canister Storage Building, that were performed for the sampling/weld station...
estimate the dose to the worker at 0.40 mrem/h to 0.60 mrem/h depending on fuel type. These shielding components are control features that prevent exposure of personnel to the high radiation of the MCO. This equipment meets the requirements for shielding as referenced in NRC equivalency criteria given in Items 20 and 23 of HNF-SD-SNF-DB-003. The carbon steel shield halves, trench cover shield, drive assembly cover shield, and center shield plate conform to ASTM A36/A36M standards.

With the shield halves in place and the center shield plate removed by the MHM, the inverted cone shape of the opening into the sampling/weld pit provides for mitigation of an accidentally dropped MCO. The thickness of the steel shield halves and the cone angle provide assurance that a dropped MCO will be guided into the sampling/weld pit, where the sampling/weld station impact absorber absorbs the energy of the dropped MCO, protecting the MCO.

A pressurized release could cause significant onsite consequences (see Chapter A3.0). The sampling/weld station design includes a pressure safety valve located on the south wall downstream of the pressure control valve and a rupture disk (PSE-1) located next to the pressure safety valve. These devices protect the sampling system and the MCO from overpressurization. The pressure safety valve is set to relieve at 135 lb/in² gauge with a tolerance of ±10%. The helium supply system rupture disk is described in Section A4.4.11. The 0.75-in. pressure safety valve is capable of discharging 20 standard ft³/min through a 10-ft-long tail piece back into the sample hood. This additional feature is provided for worker protection from high-pressure, high-temperature gas impingement in accordance with 29 CFR 1910.103 (b)(1)(ii) (29 CFR 1910).

With the MCO process valve open, a direct path exists through the MCO internal HEPA filter to the internal regions of the MCO. Actual release of contamination from the MCO is likely to be very small, and the sample hood HEPA filter (FH-009) should not become contaminated rapidly. The sample hood HEPA filter constitutes the safety-significant barrier to prevent air inleakage that would mix with the MCO gases. This filter is a sintered metal filter (316L stainless steel) contained in a stainless steel housing (304L stainless steel) and has a design pressure of 150 lb/in² with a design temperature of 400 °F. The safety-significant sample hood is designed to receive gases vented from the MCO and diluted with helium through the sample cart as well as air that enters the hood at its slotted interface with the MCO. The sample hood is a specially constructed stainless steel box with viewing windows, external lighting, and a flexible hose connection to the vent piping that leads, via the sampling/weld station gantry cranes, to the exhaust unit. The hood is suspended from the gantry crane during movement and remains connected to the crane during use. The hood incorporates code features from the following sources:

- ANSI B18.2.1, Square and Hex Bolts and Screws (Inch Series)
- ASME B30.20-1993, Below-the-Hook Lifting Devices
The hood is designed and fabricated in compliance with AWS D1.1; NFPA 70, National Electrical Code, Article 410, "Lighting Fixtures, Lampholders, Lamps and Receptacles;" NEMA 250; NFPA 255, Standard Method of Test of Surface Burning Characteristics of Building Materials; SAE J429, Mechanical and Material Requirements for Externally Threaded Fasteners; and UL 181, Factory-Made Air Ducts and Air Connectors.

The major components of the sampling/weld station MCO support structure are the rotating shield, the stationary shielding, the shear ring and shear screws, and the stationary shield support pipes (Figure A2-40). The sampling/weld station MCO support structure provides support for an MCO placed in the sampling/weld station and radiation protection for the operators. The rotating shielding carbon steel conforms to ASTM A36/A36M standards. The stationary shielding material is a laminate of borated polyethylene with carbon steel conforming to ASTM 36/A36M standards. The shear ring is made from 0.875-in.-thick ASTM A36/A36M carbon steel plate, and the alloy steel shear screws conform to ASTM F835-98, Standard Specification for Alloy Steel Socket Button and Flat Countersunk Head Cap Screws. The 6-in.-diameter, schedule 80, black carbon steel support pipes conform to ASTM A53, Grade B standards. The remainder of the sampling/weld station MCO support structure is fabricated using materials that conform to ASTM A36/A36M standards.

The sampling/weld station exhaust unit consists of a filter housing and a prefilter and is provided with a HEPA filter. The HEPA filter is a stainless steel, bag-in/bag-out style sized for a flow rate of 500 ft³/min and a face velocity of 250 ft/min. The prefilter is also a bag-in/bag-out style. The prefilter is rated at a minimum of 30% removal efficiency when tested in accordance with ASHRAE 52.1, Gravimetric and Dust-Spot Procedures for Testing Air-Cleaning Devices Used in General Ventilation for Removing Particulate Matter. The HEPA filter housing and ducting from the sample hood to the filter provide confinement of gases released from an MCO and radioactive particulate and contamination spread in accordance with ASME N509-1989 and ASME N510-1989 and meet the NRC equivalency requirement, Item 12, for HVAC systems.
The HEPA filter housing is of Type 304 stainless steel, all-welded construction in accordance with ASME N509-1989. The sampling/weld station exhaust unit is designed to meet AMCA 99, *Standards Handbook*, and AMCA 210, *Laboratory Methods of Testing Fans for Rating*, standards and is tested to ASHRAE 52.1. There is no requirement for the sampling/weld station exhaust unit to incorporate explosion-proof electrical design features because the small amount of hydrogen that could reach the fan is already diluted with helium and air. A flow indicator is included to ensure that sampling does not start if exhaust unit flow is insufficient.

The sampling/weld station stationary shielding and cooling cap are designed for removing heat from the body and top of the MCO. Insulated lines provide coolant (chilled water and propylene glycol) from the sampling/weld station chiller through pipe sleeves below the sampling/weld station operating deck to channels in the stationary shielding and to disconnect fittings and valves for the cooling cap. This system may be used if necessary to ensure that exposed MCO surfaces are sufficiently cool to permit hands-on operations such as removing the process port cover and placing the MCO process valve operator. This feature is provided for worker protection from high surface temperatures in accordance with 29 CFR 1910 and ASTM C1055-92, *Standard Guide for Heated System Surface Conditions that Produce Contact Burn Injuries*. Calculation CSB-HV-0014, *Long Term MCO Temperature Without Cooling in the Sampling Station*, assumed a steady-state outside temperature of 115 °F and a corresponding indoor temperature of 85 °F. Calculation CSB-HV-0014 determined that the performance specification surface temperature limit (270 °F) for the MCO would not be exceeded until after 30 days if an MCO were stranded in the sampling/weld station because of MIM operational problems. Therefore the cooling system is designated general service. The MCO sampling/weld station cooling system has a limited function in postrecovery cooling of the stationary shielding if an MCO is stranded. The sampling/weld station chiller is a commercially manufactured air-cooled unit sized to match the cooling needs of the sampling/weld station. It meets the requirements of ANSI/ARI 590, *Positive Displacement Compressor Water Chilling Package*; ANSI/ASHRAE 15, *Safety Code for Mechanical Refrigeration*; ANSI/ASHRAE 30, *Methods of Testing for Liquid Chilling Package*; and other appropriate standards of the American Society of Mechanical Engineers, American Society for Testing and Materials, Manufacturing Standardization Society, and the National Electrical Manufacturers Association; the unit is required to meet UL 109, *Tube Fittings for Flammable and Combustible Fluids, Refrigeration Service, and Marine Use*. The cooling cap is a custom designed and fabricated device that matches the configuration of the MCO top and cools the personnel contacted region of the MCO. Together with the piping that connects the cooling cap to the chiller unit, it is designed to meet the requirements of ANSI/ASME B31.3.

The impact to the MCO from dropped sampling/weld station shield half sections was evaluated and compared with the impact from a dropped cask lid. The analysis results are documented in Calculation CSB-RM-0016, *MCO Sampling Station—Shield Components*, and Calculation CSB-RM-0019, *Temporary Shielding Components*, and they indicate that damage to the MCO rupture disk process port cover is not sufficient to cause a release when credit is taken for the bounding MCO internal pressure of 5.2 atm. The sampling/weld station gantry crane has
the identified safety role of staying intact during a seismic event. Because the sampling/weld station hood remains connected to the sampling/weld station gantry crane during sampling operations, failure of the gantry crane could result in damage to the sampling equipment and the MCO. For this reason the gantry crane and hoist is designated ITS Category C. The seismic clamps are attached to the gantry crane and are manually engaged to the rails whenever the crane is in use over the MCO in the sampling/weld station. The seismic clamps are designed to a seismic response spectra based on 0.35 g zero-period acceleration with 5% damping for crane stability evaluation (see Figures A4-8 and A4-9).

The MCO sampling station crane is designed (and anchored as described below) to withstand the effects of a seismic event without overturning or failure. The seismic effects include the crane weight and lifted loads that result in the worst case design conditions for the crane and crane support restraint. The restraint capacity is capable of resisting the effects of an equivalent seismic acceleration acting through the center of gravity of the crane and lifted load. This equivalent acceleration is given as 1.25 g in any two orthogonal directions and 1.00 g in any vertical direction, acting simultaneously. These values were derived from peak values with 5% damping for the sampling/weld area as given in Calculation CSB-S-0040, ISRS Generation, MHM, Operating Shelter, Air Intake. The peak values, derived by SAP 90 analysis, were scaled by the ratio of 0.21/0.35 for the gantry crane and multiplied by 1.50 for equivalent static analysis per Section 10A.3.4 of the architect-engineer's Design Basis Document (Bedell 1996a).

The gantry crane is equipped with seismic restraints for the gantry in four locations; with the main hoist over the center of the pit, with the main hoist over the center of gravity of each of two sections of temporary shielding, and with the rail beam that supports the 1-ton hoists over the pit.

The gantry crane is equipped with seismic restraints for the 7.5-ton trolley at a single location when the 7.5-ton hoist is over the center of the pit. The gantry crane is designed in accordance with ANSI/ASME B30.17-1998, Overhead and Gantry Cranes (Top Running Bridge, Single Girder, Underhung Hoist), and CMAA 74-94, Specifications for Running and Underrunning Single Girder Electric Overhead Traveling Cranes Utilizing Underrunning Trolley Hoist. The underhung hoists are designed in accordance with ANSI/ASME B30.16-1998, Overhead Hoists (Underhung).

The only identified potential accidents at the sampling/weld station with the potential for significant onsite consequences are hydrogen deflagration, which is protected against by maintaining confinement of gases released from an MCO during sampling operations, and overpressurization of the MCO, which is protected against by the helium supply rupture disk (see Section A4.4.11). The sampling/weld station area steel-reinforced concrete and the adjacent walls and roof structure are contiguous with the operating deck and the operating area shelter and are designed to safety-class criteria to protect the MCOs in the storage tubes. Sampling/weld station features (including seismic restraints on the crane, and the carbon steel lower shield support columns in the sampling/weld station pits) whose failure would lead to MCO damage are designed to the seismic spectra based on 0.35 g zero-period acceleration. Structural calculations
(CSB-S-0043 and CSB-S-0026A) show that these sampling/weld station components will survive the DBE loads. The sampling/weld station support stand design has been analyzed to verify its structural adequacy for the design loads (CSB-RM-0016). The design loads for the sampling/weld station support stand include vertical loads from the stationary shield, the rotating shield, and MCO weights; vertical seismic loads; and a dropped MCO. Vertical seismic loads were not included in the analysis because a dropped MCO load is much greater. The analysis concluded that the sampling/weld station support stand design, including the rotating shield, supports the loads during normal operations and the components have adequate strength to support an MCO during a drop accident. The analysis also shows that the shear screws holding the shear ring in place have adequate strength to support an MCO and will shear when an MCO is accidentally dropped and cushioned by the sampling/weld station impact absorber. The support columns are structural features composed of 0.5-in. ASTM A36/A36M carbon steel plate and 6-in.-diameter Schedule 80 black carbon steel pipe (ASTM A53 Grade B) that provide support for the fixed shielding in the sampling/weld station pits (Figure A2-40). They are designed to ANSI/AISC N690-94, and they are welded to AWS D1.1. Pipe sleeve material and impact absorber stand are composed of carbon steel plate and shapes conforming to ASTM A36/A36M.

The sampling cart is classified general service because there are no anticipated accidents in the sampling/weld station involving the cart that are expected to have significant onsite consequences. No analyzed accidents from a DBE have been identified that would have onsite or offsite consequences.

A4.4.9.5 Controls (Technical Safety Requirements). The sampling/weld station support structure, center shield plate, trench cover shield, and drive assembly cover shield are design features (see Section A5.6), and their safety function is managed by the configuration control process.

The following assumptions associated with the MCO sampling system require TSRs to ensure performance of the safety functions.

- Overpressurization of the monitored MCOs could result in gas and particulate released to the operating area in excess of onsite guidelines. This event is precluded by having a safety-significant rupture disk in the helium supply system at the let-down station. Rupture disk PSE-1, designed to relieve at 150 lb/in², on the helium supply system to the sampling/weld station cart is credited with providing protection of the MCO from overpressurization and subsequent gaseous release. The pressure safety valves PSV-720 and PSV-728, set at 135 lb/in² gauge with a tolerance of ±10%, provide additional protection and protect the sampling/weld station components from high MCO pressure.

- The sampling/weld station shield halves are credited with mitigating the consequences of an accidentally dropped MCO and providing worker protection from excessive radiation dose. The shield halves are required to be in place for the center shield plate to be at the MHM plug grapple height. The MHM interlocks
discussed in Section A4.4.4 prevent placement of the MCO unless the shielding surface is in place. To provide significant worker safety and significant defense in depth, Administrative Control (AC) 5.9, described in Chapter A5.0, contains requirements to verify that the sampling/weld station shield halves have been installed before sampling operations are begun. After the MHM has placed an MCO into the sampling/weld station and has returned to its staging position, the sampling/weld station gantry crane removes the center shield plate and the shield halves and places the sample hood on the rotating shield.

- The sample hood and the sampling/weld station exhaust system are designed to dilute the potential hydrogen releases to below concentrations that would be flammable. Just before sampling operations are begun, the following inspections are performed:

  - Sample hood glove ports for degradation
  - Flexible hose and connections for integrity
  - Balance and isolation dampers MD-5, MD-6, and MD-8 full open; MD-7 and MD-9 partially open.

Support systems required for sampling operations include 480 V power to the combination starter for the exhaust fan, to the disconnect switch for the sampling/weld station cooler, to the motorized gantry and hoist, and to the sampling/weld station rotation motor; 120 V power to the pyrometer panel and receptacles.

After placing the sample hood and establishing that all equipment is operable, verify flow rate through the hood of 100 ft³/min minimum by observing the sampling/weld station HVAC flow indicator and the sample hood pressure differential indicator.

- Leak testing of the MCO process valve operator and connection to the sample cart is credited with preventing releases of potentially flammable MCO gases into the sample hood space. During a loss of power, this combination of MCO gas leak and loss of dilution airflow through the hood could result in the accumulation of a flammable concentration of hydrogen gas. Before sampling operations are begun, the MCO process port connection and attached flexible hose piping for the sample system are leak tested to ensure that no flammable atmosphere could be created inside the sample hood. Leak test of the sample line to the MCO process port valve operator shall be shown to be less than 40 cm³/s at 75 lb/in² gauge (as confirmed by a pressure decay rate). Verification of this test is required according to Chapter A5.0, LCO 3.2.2.
A4.4.10 Standard Storage Tube Intermediate Impact Absorber

A4.4.10.1 Safety Function. The safety function of the intermediate impact absorber is to protect both the bottom MCO and the accidentally dropped MCO from damage exceeding the design requirements. In the absence of an intermediate impact absorber, the dropped MCO causes the welded cap on the MCO to bulge downwards and outwards. This could make it difficult to retrieve the bottom MCO and could cause significant damage to the dropped MCO (see Figure A4-10).

A4.4.10.2 System Description. The intermediate impact absorbers are described in Section A2.4.3 and are shown on Figure A2-13, sheet 2. The intermediate impact absorbers (see Figure A2-13) consist of the following components: a flange plate with a top plate bolted to the flange plate, the base plate, and energy-absorbing tube components in the middle. The flange plate with the top plate is 3.01-in.-thick carbon steel material conforming to ASTM A36/A36M. The base plate is constructed of 1.5-in. carbon steel plate conforming to ASTM A36/A36M. Carbon steel material conforming to ASTM A513, Standard Specification for Electric-Resistance-Welded Carbon and Alloy Steel Mechanical Tubing, is selected for the tube sections. The maximum diameter of the lower impact absorbers is 26 in. The overall height is 25.35 in. after factory precrush of the assembled intermediate impact absorber. The top and bottom plates serve to maintain the impact absorber centered in the storage tube. Galvanized steel cables spaced equidistantly around the absorber circumference maintain the pipe components in place.

A4.4.10.3 Functional Requirements. Limiting the energy imparted by the dropped MCO to the bottom MCO to below 35 g has been specified as a criterion for the intermediate impact absorbers. This value is provided in HNF-S-0426, Performance Specification for Spent Nuclear Fuel Project, Multi-Canister Overpack, for the MCO. The MCO has been designed to withstand the stresses associated with a 35 g loading (HNF-SD-SNF-SARR-005).

A4.4.10.4 System Evaluation. Intermediate impact absorbers are designed to protect the MCO during the accidental drop event. The vendor is required to perform testing on samples of the materials before fabricating each lot of intermediate impact absorbers. The test results will be used to confirm that the intermediate impact absorber will mitigate damage to an accidentally dropped MCO and to a lower capped MCO. A drop analysis for an MCO accidentally falling onto an MCO stored in a standard storage tube estimates values of 28 g and 16 g, respectively, for the falling MCO without an intermediate impact absorber and with an intermediate impact absorber. See SNF-4042, Evaluation of Accident Frequencies at the Canister Storage Building, and SNF-3328 for additional details. The analysis shows that damage to the falling MCO and the bottom MCO also is mitigated by the bottom impact absorber. Both of these values are within the design limits for an MCO. Without an intermediate impact absorber in place, an MCO dropped on top of the bottom MCO (Figure A4-10) would deform the canister cover assembly of the bottom MCO and impair retrievability. With an intermediate impact absorber in place, deformation of the bottom MCO canister cover assembly would not occur because the bottom impact absorber and the intermediate impact absorber are designed to protect the bottom and top MCOs.
A4.4.10.5 Controls (Technical Safety Requirements). The following assumption associated
with the intermediate impact absorber requires TSRs to ensure performance of the safety function.
This is controlled by administrative procedures discussed in Chapter A5.0 under AC 5.10.

- The intermediate impact absorber must be verified to be in place by the MHM
  television camera before a second MCO is put into the tube.

A4.4.11 Helium Supply System Rupture Disk

A4.4.11.1 Safety Function. The safety function of the rupture disk (PSE-1) on the helium
supply system to the sampling/weld station is to prevent damage to MCOs caused by
overpressurization with helium. Design pressure limits for the MCO are given in the MCO
Topical Report (HNF-SD-SNF-SARR-005). Helium supply system is designed for a pressure of
150 lb/in² gauge and temperatures up to 270 °F. Exceeding this design pressure limit could result
in internal changes severe enough to prevent MCO process functions, exceed load limits specified
in the MCO performance specifications, or possibly fail confinement (see Section A4.3.7).

Overpressurization of the MCO through inadvertent charging with high-pressure helium
(2,000 lb/in² gauge) has been identified as a potential accident that must be precluded by safety-
significant design features as determined by Section A3.4.2.2, “Gaseous Release from the
Multi-Canister Overpack.” This event is precluded by a rupture disk (PSE-1) on the helium
supply let-down station. This device (PSE-1) is designated safety significant in accordance with
DOE-STD-3009-94 and Letter 97-SFD-172 (Sellers 1997) and ITS Category B using the graded
approach of HNF-SD-SNF-DB-003. If a helium overpressurization condition occurs, PSE-1
vents into the operating area. The helium mixes with the large air volume being circulated by the
ventilation system and does not create an oxygen-deficiency problem for the operations personnel.
Pressure safety valves on the sample line designated general service (PSV-720 and PSV-728)
provide overpressurization protection for sampling/weld station equipment whose unmitigated
failure would result in releases.

A4.4.11.2 System Description. The helium supply to the sampling/weld station is provided by
an extension of the 2-in. line supplying the helium refill station to the south and east of the helium
let-down station (see Section A2.7.5 and Figure A2-42). Pressure control valve PCV-733
reduces the helium supply line pressure from 2,000 lb/in² gauge to 120 lb/in² gauge. Line size for
PCV-733 is 0.5 in. Line size for PSE-1 is 1.0 in. The reduced pressure helium supply is delivered
to the sampling/weld station via 0.75-in. lines that are routed below the deck to connection boxes
in pits 3 and 6, which adjoin the sampling/weld station pits 2 and 7. Overpressurization
protection of the sampling/weld station supply line and sample cart components is also provided
by PSV-728. Protection of sampling/weld station components from excessive MCO pressures is
provided by PSV-720 located downstream of the canister filter as part of the sampling/weld
station hood.
A4.4.11.3 Functional Requirements. The helium rupture disk must be adequately sized to prevent the 2,000 lb/in² gauge helium supply from pressurizing the MCO beyond its 150 lb/in² gauge design limit. PSV-720 is subject to the temperatures and gas composition from the MCO interior and must be designed to function in that temperature environment. The likelihood of a seismic event occurring in concert with sampling operations has been determined to be beyond extremely unlikely. Therefore, failure of pressure relief devices in the closed position during a seismic event has been excluded as a design requirement.

A4.4.11.4 System Evaluation. Pressure safety devices PSV-728 and PSE-1 are mounted on the south wall of the sampling/weld station above the curb at elevation 714 ft and are designed to relieve helium overpressurization through a 10-ft-long tail piece sized by the valve vendor. This ensures that workers would not be exposed to high noise levels in the unlikely event of a valve or rupture disk actuation. Pressure safety relief device PSE-1 is specified to relieve at 150 lb/in² gauge. PSV-728 is sized for the case in which pressure control valve PCV-733 fails wide open with pressures of 2,000 lb/in² gauge upstream and 150 lb/in² gauge downstream of PCV-733. Pressure safety device PSV-720 is sized to relieve 20 standard ft³/min at temperatures up to 400 °F. Discharge from PSV-720 is routed back to the sampling/weld station hood for worker protection. Pressure safety device PSV-728 setpoint for overpressurization is 135 lb/in² gauge ±10% to allow for margin below the MCO design pressure. Pressure safety devices PSV-728 and PSE-1 are also specified to be furnished in accordance with ASME B16.34-1996, *Valves—Flanged, Threaded, and Welding End*. They have carbon steel bodies and 316 stainless steel trim.

A4.4.11.5 Controls (Technical Safety Requirements). The rupture disk PSE-1 is a design feature (see Section A5.6) and its safety function is managed by the configuration control process.

A4.4.12 Multi-Canister Overpack Handling Machine Fixed Shielding System

A4.4.12.1 Safety Function. The safety function of the MHM fixed shielding system (including the shield skirt) is to protect workers from radiation exposure from the MCO. This criterion is met by reducing the maximum total dose rate at a 30-cm distance from any exterior location on the MHM to less than 0.5 mrem/h. The MHM fixed shielding system provides significant worker safety for personnel located in the operating area and is therefore designated a safety-significant design feature of the MHM.

A4.4.12.2 System Description. The MHM is fully shielded for gamma and neutron radiation emitted from an MCO. The gamma shielding of the main cask body, which encloses the MCO while in transit, is manufactured from cast steel. This cask body was manufactured in sections that are rigidly bolted together using high tensile studs and face-to-face contact joints. The horizontal interface gap between the base of the rotating turret and the top face of the stationary nose casting is 0.08 in. thick and is stepped to prevent direct radiation streaming from the gap. Above the level of the MCO in the cask, a steel slab integral with the grapple assembly and the
MCO hoist confinement chamber walls provide gamma protection. The neutron shielding around the turret is provided primarily by 4-in.-thick Jabroc N, a densified wood product.

The nonrotating nose also is manufactured from cast steel and, in conjunction with the components of the retractable nose, has a net shielding thickness comparable to that of the main cask body. A circular slab forming part of the nose section provides gamma attenuation when the cask is rotated from the "nose aligned" condition during intermediate operations and when the television camera is aligned over the nose cavity. A circular steel slab at the bottom end of the rotating turret provides protection from gamma radiation from an MCO in the vault when the MHM is located over an open storage tube. At the turntable fabrication and around the fixed nose and retractable shield skirt, the neutron shielding is provided by cast concrete sections.

**A4.4.12.3 Functional Requirements.** The MHM is provided with interlocks to prevent the MCO from being lowered if the MHM shield skirt is not in proper position to shield the worker. The shield skirt and the MHM fixed shielding constitute control devices that ensure exposure to the worker from the MCO (which emits radiation from its lateral surfaces at greater than 100 mrem/h at 30 cm) is in compliance with the requirements of Title 10, *Code of Federal Regulations*, Part 20, "Standards for Protection Against Radiation," Section 20.1601, "Control of Access to High Radiation Areas" (10 CFR 20), as referenced in HNF-SD-SNF-DB-003, Item 20. The fixed and moveable MHM shielding protects the operator or other facility worker from excessive doses of radiation in keeping with guidance from NRC Regulatory Guide 8.8, *Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as Reasonably Achievable*, as referenced in HNF-SD-SNF-DB-003, Item 23. The MHM service pit provides a location with equipment that can be used without resulting in excessive exposure. Maintenance activities at the MHM service pit are not planned to be conducted when an MCO is contained within the MHM. HNF-SD-SNF-DB-003, Item 20, also imposes the requirement of 10 CFR 20, Section 1301, "Dose Limits for Individual Members of the Public," that the dose rate to the public is limited to 2 mrem/h for any unrestricted area from external sources during normal operations.

**A4.4.12.4 System Evaluation.** Interlock P2 is considered safety significant due to the possible dose impact on the facility worker of not replacing a tube plug in the storage tube before moving the MHM. The MHM fixed shielding system has been shown to adequately reduce the maximum total dose rates around the MHM when carrying a bounding radiation-emitting MCO. The maximum calculated dose rate 30 cm from an outer MHM surface was 0.28 mrem/h (0.0028 mSv/h) [ESL/R(96)085]. This maximum dose rate permits personnel to work around the MHM 40 hours a week all year without exceeding any Hanford Site total dose administrative limits. During normal operations and anticipated events in the CSB, the 2 mrem/h criterion of 10 CFR 20.1301 will not be exceeded.

The MHM shield skirt, when fully lowered, has been shown to adequately reduce the maximum total dose rate around the MHM as a bounding radiation-emitting MCO is being lowered from the MHM. In ESL/R(96)085, *Hanford MHM Phase III Shielding Assessment*, the
maximum dose rate 30 cm from an outer MHM surface was calculated to be 0.28 mrem/h, which
was less than the target value of 0.5 mrem/h.

A4.4.12.5 Controls (Technical Safety Requirements). The MHM fixed shielding system is a
design feature of the MHM (see Section A5.6), and its safety function is managed by the
configuration control process.

A4.4.13 Transportation Cask Shielding

A4.4.13.1 through A4.4.13.4. The safety function of transportation cask shielding is to protect
the facility operator from excessive radiation dose. The MCO transportation cask shielding has
been classified safety significant. The MCO transportation cask safety analysis report for
packaging (HNF-SD-TP-SARP-017) demonstrates that the MCO transportation cask meets the
onsite transportation safety criteria. The cask shielding is an integral part of the MCO
transportation cask, is required for worker safety protection, and is classified as safety significant
in Chapter A3.0. The transportation cask shielding is a control device that protects the CSB
facility worker from radiation doses from the high radiation area internal to the cask containing an
MCO in compliance with the requirements of 10 CFR 20.1601 as cited in HNF-SD-SNF-DB-003,
Item 20. At the CSB there are no normal or anticipated events involving the shielded cask that
result in exposure to the public in excess of the 10 CFR 20.1301 dose limit of 2 mrem/h, also
cited in HNF-SD-SNF-DB-003, Item 20.

A4.4.13.5 Controls (Technical Safety Requirements). Transportation cask shielding is a design
feature (see Section A5.6), and its safety function is managed by the configuration control
process.

A4.4.14 Multi-Canister Overpack Shield Plug Shielding

A4.4.14.1 through A4.4.14.4. The safety function of the MCO shield plug shielding is to protect
the facility operator from excessive radiation dose. This is done by reducing the maximum dose
rate at contact (<10 cm distance) from the top of the MCO shield plug to an average of 2 mrem/h
and not greater than 8 mrem/h. See the MCO Topical Report (HNF-SD-SNF-SARR-005) for the
safety functions, system description, functional requirements, and system evaluation of the MCO
shield plug shielding. The MCO shield plug is an integral part of the MCO, is required for worker
safety protection, and is classified as safety significant in Chapter A3.0. The MCO shield plug
provides protection of the worker from excessive dose in compliance with the requirements of
10 CFR 20.1601 that control devices be in place for all high radiation areas (the inside of the
MCO is a high radiation area). There are no requirements to remove the MCO shield plug after it
is mated to the rest of the MCO. Workers will attach the sample hood to the process valve port 2
on the MCO top when taking a sample from a monitored MCO. Additional shielding is in place at
that time to protect them from excessive exposure. It should also be noted that only very small
amounts of radioactive material are expected to travel from the MCO into the sample cart piping.
because of the HEPA filter FH-009 in the sample hood and the internal HEPA filter inside the MCO shield plug at port 2. Therefore, the requirements of NRC Regulatory Guide 8.8 as referenced in HNF-SD-SNF-DB-003 are met in that features are provided to minimize personnel doses associated with the MCO shield plug.

At the CSB there are no normal or anticipated events involving the MCO shield plug that result in exposure to the public in excess of the 10 CFR 20.1301 dose limit of 2 mrem/h, also cited in HNF-SD-SNF-DB-003, Item 20.

A4.4.14.5 Controls (Technical Safety Requirements). MCO shield plug shielding is a design feature (see Section A5.6), and its safety function is managed by the configuration control process.

A4.4.15 Standard Interface Guide Ring Funnel, Storage Tube Lower Flanges, and Bottom Impact Absorbers

A4.4.15.1 Safety Function. The standard interface guide ring funnel, storage tube lower flanges, and bottom impact absorbers perform the safety function of protecting the MCO during the potential drop accident. The standard interface guide ring funnel and storage tube lower flange mitigate the effects of a potentially dropped MCO by guiding the MCO into the storage tube where it is cushioned by the bottom impact absorber. Bottom impact absorbers serve to reduce the deceleration forces on a dropped MCO to below the 35 g limit found in the MCO Topical Report (HNF-SD-SNF-SARR-005).

A4.4.15.2 System Description. The standard interface guide ring funnel, storage tube lower flange, and bottom impact absorber are shown in Figures A2-8, A2-12, A2-13, and A2-27. The lower flange is part of the standard storage tube assembly welded between the bellows and the tube. The lower flange is made from 4.5-in.-thick steel plate conforming to ASTM A36/A36M with an inverted cone opening having a 45° seating surface, matching the seating surface of the standard storage tube plug. The lower flange outer diameter is 39 in. with an inner diameter opening of 27 in.

The interface guide ring funnel is a spacer placed between the storage tube plug and the MHM nose piece (Figure A2-27). The interface guide ring funnel is moved and placed using the tube vent and purge cart hoist. The interface guide ring funnel fits snugly into the top of the deck embed, resting on the bellows assembly top flange (note operations described in Section A2.5.1.3). The interface guide ring funnel provides a resting surface for the MHM nose piece’s machined surfaces. The interface guide ring funnel is fabricated from ASTM A36/A36M carbon steel using AWS D1.1 as the welding code and 100% visual examination in accordance with Paragraph 6.9 of D1.1-96. The extension piece consists of 0.5-in.-thick plate sections constructed of carbon steel conforming to ASTM A36/A36M. Lateral movement of the interface guide ring funnel is restricted by the deck embed and the tube assembly.
Bottom impact absorbers are provided at the base of all standard storage tubes. The bottom impact absorbers mitigate the effects of an MCO drop to limit MCO impact forces and prevent possible rearrangement of the MCO internals. The bottom impact absorbers (see Figure A2-13) consist of the top flange plate, a top plate bolted to the flange plate, an intermediate plate, the base plate cable assemblies, and energy-absorbing pipe components between the plates. The flange plate is 2.13-in.-thick carbon steel material conforming to ASTM A36/A36M. The top, base, and intermediate plates are constructed of 0.88, 0.5, and 0.88-in. carbon steel plates, respectively, conforming to ASTM A36/A36M. Carbon steel material conforming to ASTM A513, Type 5, is selected for the tube sections. The maximum diameter of the lower impact absorbers is 26 in. The overall height is about 34.65 in. after factory precrush of the assembled bottom impact absorber. The top and bottom plates serve to maintain the impact absorber centered in the storage tube. Galvanized steel cables spaced equidistantly around the absorber circumference maintain the tube components in place.

The impact absorbers are designed to preclude overstressing the MCO by absorbing the kinetic energy of a free-falling MCO and safely decelerating it. That is, the tube components between the top, intermediate, and bottom plates are crushed by the MCO, absorb the kinetic energy, and prevent the MCO from striking the bottom and becoming overstressed. Preliminary testing has demonstrated that material strength, operating temperature, number of tubes, tube thickness, precrimping the lower part of the tube (precrushing, installing short and long tubes), and crush stroke length are important parameters in designing the impact absorbers (PacTec 1999). These parameters provide for predictable results and a smooth force-deflection curve, and under impact conditions, the drop testing results provide assurance that the impact absorbers will preclude overstressing the MCO. After receipt of the tube stock, a preselected number of impact absorbers will be assembled. Full-scale testing of a preselected number of assembled bottom impact absorbers and cask receiving impact absorbers will determine the number of tubes needed in each impact absorber to prevent overstressing of the MCO. This method of receiving a batch of tubes, assembling the test impact absorbers, and full-scale testing will determine the number of tubes required for the bottom impact absorbers, the intermediate impact absorbers, the cask receiving impact absorber, and the sampling/weld station impact absorbers.

**A4.4.15.3 Functional Requirements.** A drop of an MCO onto the edge of a storage tube is mitigated by the 45° angle of the lower flange of the standard storage tube. The bottom impact absorber protects the MCO from exceeding its specified design limits during an accidental drop from heights up to 44 ft by limiting the total force (maximum imposed force on the MCO is 679,000 lb-force) transmitted to the MCO. The total force transmitted is not to exceed 680,000 lb-force for an MCO weighing 20,000 lb dropped from a height of 44 ft. The bottom impact absorber also protects the storage tube from exceeding the pipe wall material working strength of 750 lb/in² during an MCO drop. Design life of the bottom impact absorber with the weight of two MCOs and an intermediate impact absorber is specified to be 40 years (with an expected extension to 75 years) at the design temperatures and radiation levels anticipated without settling. Design temperature for the standard storage tubes is 220 °F.
The interface guide ring funnel is required to withstand the eccentric drop of an MCO while guiding the MCO into the storage tube. Damage or deformation to the interface guide ring funnel is acceptable as long as the MCO is guided from the eccentric drop position to the analyzed configuration of MCO contact with the lower flange.

A4.4.15.4 System Evaluation. The thickness of the lower flange and the angle of the sealing surface provide assurance that a dropped MCO will be guided into the storage tube and cushioned by the impact absorber. A preselected number of storage tube bottom impact absorbers and cask receiving impact absorbers will be drop-tested to determine the number of tubes needed to protect the MCO during a drop accident and to ensure that neither MCO loading due to deceleration nor any tube wall contact pressures exceed the design limits. The impact absorbers are custom-designed by the supplier to specified crush strength and dimensional envelope requirements. The structural safety-class code for the analysis, design, and fabrication of the storage tube components including impact absorber is ANSI/AISC N690-94. Final design configuration for each tube batch will be based on the drop test results. Loads on the supporting structure for the storage tube bottom impact absorber were considered in calculation CSB-S-0007, Appendix 6. The results indicate that a concentric drop of an MCO with no impact absorber in place will not crush or damage the base assembly or the basemat embeds or exceed the basemat's punching shear capacity.

Calculation CSB-S-0062, MCO Drop on Guide Ring Flange, evaluated the accidental eccentric drop of an MCO from the MHM onto the storage tube and the effect of the drop on the interface guide ring funnel. The analysis assumed that the interface guide ring funnel would be in position and deflect the dropped MCO onto the lower bellows flange. The results of the analysis for impact indicate that the dropped MCO is guided into the storage tube without much contact with the bellows lower flange. The MCO transmits approximately 15% of its kinetic energy into the interface guide ring funnel. The interface guide ring funnel sustains some damage (about 46% limited to a very small area), but it is able to perform its safety function during the postulated accident event. Impact to the concrete reinforced deck was about 2,720 lb/in², and impact to the carbon steel deck embed flange upon which the interface guide ring funnel rests was about 17,500 lb/in².

A4.4.15.5 Controls (Technical Safety Requirements). The standard storage tube lower flange is a design feature and its safety function is managed by the configuration control process (see Section A5.6). The following assumptions associated with the standard storage tubes, interface guide ring funnel, bottom impact absorbers, and tube base assemblies require TSRs to ensure performance of the safety functions.

- The bottom impact absorber is in place in each standard tube before positioning the first MCO (see AC 5.10).
- Verify that the standard interface guide ring funnel is in place before placing an MCO or removing an MCO from a storage tube.
A4.4.16 Overpack Interface Guide Ring Funnel, Storage Tube Lower Flanges, and Bottom Impact Absorbers

A4.4.16.1 Safety Function. The overpack interface guide ring funnel, storage tube lower flanges, and bottom impact absorbers perform the safety function of protecting the MCO during the potential drop accident. The overpack interface guide ring funnel and storage tube lower flange mitigate the effects of a potentially dropped MCO by guiding the MCO into the overpack storage tube where it is cushioned by the bottom impact absorber. The overpack storage tube is provided with an impact absorber similar to the standard storage tube impact absorber (see Section A4.4.15). Bottom impact absorbers serve to reduce the deceleration forces on a dropped MCO to below the 35 g limit found in the MCO Topical Report (see HNF-SD-SNF-SARR-005).

A4.4.16.2 System Description. The overpack interface guide ring funnel, storage tube lower flange, and bottom impact absorber are shown in Figures A2-8, A2-12, A2-13, and A2-27. The lower flange is part of the overpack storage tube assembly welded between the bellows and the tube. The lower flange is made from 5.5-in.-thick steel plate conforming to ASME SA516 (ASTM A516/A516M), GR-70, with an inverted cone opening having a 45° seating surface, matching the seating surface of the overpack storage tube plug. The lower flange outer diameter is about 45.25 in. with an inner diameter opening of 27 in. The overpack interface guide ring funnel is nearly identical to the standard interface guide ring funnel described in Section A4.4.15. The overpack storage tube bottom impact absorber is identical to the impact absorber described in Section A4.4.15.

A4.4.16.3 Functional Requirements. A drop of an MCO onto the edge of an overpack storage tube is mitigated by the 45° angle of the lower flange of the overpack storage tube. The overpack interface guide ring funnel and bottom impact absorber protect the MCO from exceeding its specified design limits during an accidental drop as described in Section A4.4.15.

A4.4.16.4 System Evaluation. The thickness of the lower flange and the angle of the sealing surface provide assurance that a dropped MCO will be guided into the overpack storage tube and cushioned by the impact absorber. The results of the analysis of an eccentric drop of the tube plug from the MHM (an 87.75-in. maximum drop height) are documented in Calculation CSB-S-0061, MCO and Plug Drop Effects Analysis, Section 6.2. The results indicate that a significant amount of local damage occurs to the contacted portion of the standard tube flange. The results indicate that the tube plug can be reinserted and the MCO placed or removed following appropriate recovery actions to restore the MHM’s functionality. The reinstalled plug may not completely seal along the flange-to-plug sealing surface because of residual flange deformations. As a minimum, the overpack storage tube plug should be exchanged for one with undamaged sealing surfaces. An eccentric dropped plug onto an overpack storage tube assembly removes this storage tube from consideration as a suspect MCO storage tube. The tube’s flange sealing surface is expected to be damaged, and its ability to seal is questionable. As a minimum, the plug could be exchanged for one with undamaged sealing surfaces. Preferably, the MCO could be placed in an adjoining overpack storage tube.
Calculation CSB-S-0062 evaluated the accidental eccentric drop of an MCO from the MHM onto the storage tube (see Section A4.4.15). Results of the analysis of a concentric drop of a tube plug onto the standard tube lower bellows flange (a 126-in. maximum drop height) are documented in Calculation CSB-S-0050, MCO and Plug Drop Effects of the Shimmed Storage Tube Assembly and Related Components, Appendix D. The results indicate that negligible permanent damage occurs. The resultant net decrease in the inner diameter of the standard tube was determined to be negligible. It is expected that the concentric drop of a tube plug onto the overpack storage tube’s lower bellows flange will not affect tube plug or MCO placement. As a minimum, the dropped tube plug will be tested to ensure that the seals can contain an inert gas pressure in the overpack storage tube and the locking mechanism can lock an overpack storage tube plug into position. The overpack storage tube plug will be replaced or repaired if the tests show that the overpack storage tube plug cannot hold an inert gas pressure in the overpack storage tube.

A4.4.16.5 Controls (Technical Safety Requirements). The overpack storage tube lower flange is a design feature and its safety function is managed by the configuration control process (see Section A5.6). There are no TSR controls for this SSC because its safety function is related to recovery actions, not to accidents during normal operations. Controls related to specific recovery actions must be developed on a case-by-case basis depending on the current conditions and recovery evaluation results.

A4.4.17 Cask Lifting Yoke

A4.4.17.1 Safety Function. The receiving crane is used each time a transportation cask containing an MCO is removed from the transporter and placed in the cask receiving pit in the load-in/load-out area of the CSB. The safety-significant safety function of the cask lifting yoke is to prevent lifting a cask-MCO to a height from which it would sustain unacceptable damage if dropped. The cask lifting yoke is a design feature that is credited with preventing the drop height from being exceeded. The cask lifting yoke is designated safety significant in accordance with DOE-STD-3009-94 and Letter 97-SFD-172 (Sellers 1997).

A4.4.17.2 System Description. Chapter A2.0, Section A2.5.1.1.1, and Figure A2-21 contain a description of the cask lifting yoke. The cask lifting yoke, conforming to ASTM A36/A36M, has two lower hooks that interface with the transportation cask trunnions and an upper bar that interfaces with the receiving crane hook. An emergency stop push button installed near the receiving crane can be used to stop the receiving crane to prevent travel over the FFTF pit or the MHM maintenance pit. This push button is in series with the shunt activation contact in the interface panel.

A4.4.17.3 Functional Requirements. Safe handling of the MCO transportation cask is achieved by ensuring that the receiving crane will not raise the cask above 60 in. from the floor.
The safety-significant protective feature provided by the cask lifting yoke is required to provide a high level of assurance that the receiving crane main hoist will not raise the transportation cask above the 60-in.-height limit required by Section A3.4.2.1. Safe handling of the cask-MCO ensures that the cask-MCO will not be dropped and damaged.

**A4.4.17.4 System Evaluation.** Analysis and review of the cask and trunnion in HNF-SD-SNF-FDR-003 and an analysis of the receiving crane lifting yoke in HNF-SD-SNF-DA-022 demonstrate that the cask lifting equipment complies with ANSI N14.6-1993 requirements.

In HNF-3243, *Detailed Simulation of Cask and Multi-Canister Overpack Vertical Drop onto the Canister Storage Building Receiving Area Floor*, a drop of the cask-MCO from a height of 60 in. onto the concrete floor of the CSB receiving area is shown to result in no releases of radioactivity.

**A4.4.17.5 Controls (Technical Safety Requirements).** The following assumptions associated with the cask lifting yoke require TSRs to ensure performance of the safety function.

- Verify, before using the receiving crane to lift a transportation cask, that the correct lifting yoke for the cask is used to ensure that the 60-in. height limit is not violated (see Section A5.5.3.3 and A5.6.15).

- Perform routine and recommended surveillance and nondestructive examination on the stressed components of the lifting yoke as recommended by the vendor or ANSI N14.6-1993, whichever is more restrictive. ANSI N14.6-1993 requires initial and annual acceptance testing at 150% of the maximum service load. These tests are followed by visual examination. Alternatively, in areas where surface cleanliness and conditions permit, dimensional testing, visual inspection, and nondestructive testing shall suffice.

**A4.4.18 Multi-Canister Overpack Centering Guide**

**A4.4.18.1 Safety Function.** The safety function of the MCO centering guide, located in the MHM MCO cavity, is to ensure that an accidentally dropped MCO will always impact on the inclined surfaces of the cask receiving pit shield hatch and MCO guide assembly, the standard or overpack storage tube lower flange, or the sampling/weld station shield halves. This MCO cavity feature protects the MCO from damage. The MCO centering guide is designated safety significant in accordance with DOE-STD-3009-94 and 97-SFD-172 (Sellers 1997).

**A4.4.18.2 System Description.** The MCO centering guide is a spacer attached to the inner wall of the MHM MCO cavity (Figure A2-33). The MCO centering guide fits snugly inside the MHM MCO cavity and is bolted to the cavity wall. The MCO centering guide is fabricated of carbon steel conforming to ASTM A36/A36M. The MCO centering guide positions an MCO suspended...
by the MCO grapple in the middle of the MCO cavity and prevents the MCO from drifting too far from the center of the MCO cavity.

A4.4.18.3 Functional Requirements. The MCO centering guide is required to position the MCO in the center of the MHM MCO cavity to prevent a dropped MCO from hitting the edge of shields or flanges when the MCO is lowered. The MCO centering guide prevents an eccentric drop from striking an edge of the shield hatch and MCO guide assembly, the storage tube lower flange, or a sampling/weld shield half. Damage or deformation of an MCO is minimized when an MCO strikes the inclined surfaces of these shields or flanges.

A4.4.18.4 System Evaluation. The accidental eccentric drop of an MCO from the MHM has been evaluated and is summarized in Sections A4.4.15, A4.4.16, A4.4.19, and A4.4.20. The results of the analyses show some damage to the inverted cone surfaces leading into the storage tube, the sampling/weld station pit, the cask receiving pit with deflection of the dropped MCO into the cavity and absorption of the fall by the impact absorbers. The MCO centering guide prevents the dropped MCO from striking the edges of the inverted cone surfaces.

A4.4.18.5 Controls (Technical Safety Requirements). The MCO centering guide is a design feature of the MHM (see Section A5.5.3.3), and its safety function is managed by the configuration control process. There are no special maintenance or surveillance requirements.

- Verify that the MCO centering guide is in place before the MHM leaves the maintenance pit in the MCO mode.

A4.4.19 Cask Receiving Impact Absorber

A4.4.19.1 Safety Function. The cask receiving impact absorber has the safety function to protect the cask–MCO dropped by the receiving crane into the cask receiving pit from unacceptable damage. The cask receiving pit impact absorber is designated safety significant because of its role in limiting the deceleration forces to the MCO.

A4.4.19.2 System Description. The cask receiving impact absorber, placed in the cask receiving pit, cushions the cask–MCO if it is accidentally dropped and supports the cask–MCO at floor level making the cask lid accessible for cask lid bolt loosening and cask lid removal operations. The impact absorber positions the top of the MCO at the proper height for the MHM to retrieve it from the transportation cask. A shield hatch and MCO guide assembly (Figure A2-17) placed over the cask receiving pit at deck level provide radiation protection for the facility worker (see Section A4.4.7) and interfaces with the MHM (see Figure A2-32).

The cask receiving impact absorber (see Figure A2-9, sheet 5) consists of the following components: the top plate, the base weldment, and energy absorbing tube components between the top plate and base weldment. The top plate is 1.5-in.-thick carbon steel material conforming to ASTM A36/A36M. The base weldment is constructed of carbon steel pipes and plates.
conforming to ASTM A53 and ASTM A36/A36M. Carbon steel material conforming to
ASTM A513, Type 5, is selected for the tube sections. The maximum diameter of the cask
receiving impact absorber is 53 in. The overall height is approximately 44 in. after factory
precush of the assembled cask receiving impact absorber. The base weldment serves to maintain
the impact absorber centered in the cask receiving pit. Alloy steel rods spaced equidistantly
around the absorber circumference maintain the pipe components in place. These rods conform to
ASTM A193/A193M, Standard Specification for Alloy-Steel Stainless Steel Bolting Materials
for High-Temperature Service.

A4.4.19.3 Functional Requirements. The cask receiving impact absorber must absorb the
energy from a dropped cask-MCO to the extent that deceleration forces to the MCO are below
the specified design limits. The MCO’s DBA loading limits are provided in the MCO Topical
Report (HNF-SD-SNF-SARR-005). The MCO is required to survive deceleration loads up to
35 g. According to HNF-SD-SNF-DP-007, Multi-Canister Overpack/Cask Drop Analysis File
Documentation, Appendix O, the dropped cask-MCO has been found to exceed the 35 g
specification limits of the MCO without an impact absorber in the cask receiving pit. The cask
receiving impact absorber is specified to limit the dropped cask-MCO forces to the MCO from
the maximum drop height of 60 in. above the operating deck into the cask receiving pit to less
than 35 g. The specified loaded cask weight is 59,000 lb. The loaded cask can be dropped from a
maximum height of 23 ft, 5 in. onto the cask receiving impact absorber without exceeding the
35 g specification limit.

A4.4.19.4 System Evaluation. Analyses of a dropped loaded cask-MCO into the cask receiving
pit without an impact absorber are documented in Calculation CSB-S-0014, Deck Cantilever/
Load-In/Load-Out Design. The analyses show that the drop penetration depth using the
Modified National Defense Research Committee Formula requires a slab thickness greater than
that provided. For a 19-ft unmitigated drop, the pit foundation base slab required is 29.3 in. as
compared to the 24 in. provided. No structural consequences other than damage to the pit are
caused by this unmitigated drop scenario. The functional requirements listed above are provided
in performance specifications to the supplier. The specification requirements include conformance
with ANSI/AISC N690-94 for design and fabrication. Specification requirements include that a
prototypical drop test be performed. Before the cask receiving impact absorber is installed in the
cask receiving pit, vendor testing will confirm that the cask receiving impact absorber will
mitigate damage to an accidentally dropped cask-MCO.

A4.4.19.5 Controls (Technical Safety Requirements). The cask receiving pit guides are a
design feature of the MCO cask receiving pit, and their safety function is managed by the
configuration control process (see Section A5.6).

The following assumption associated with the cask receiving impact absorber requires a
TSR to ensure performance of the safety functions (see Section A5.5.3.4).

- Verification that the cask receiving impact absorber is in place before facility startup
  and following any required maintenance on the cask receiving impact absorber.
A4.4.20 Sampling/Weld Station Impact Absorber and Shield Halves

A4.4.20.1 Safety Function. The shield halves and impact absorber at the sampling/weld station have the safety function to protect an MCO dropped by the receiving crane into the sampling/weld station from unacceptable damage. The sampling/weld station shield halves, and impact absorber are designated safety significant because of their role in reducing the impact force and limiting the deceleration forces to the MCO.

A4.4.20.2 System Description. The rotating shield supports the MCO below the operating deck at the sampling/weld station. A sampling/weld station impact absorber is located below the MCO. The shield halves and center shield plate (Figure A2-40) placed over the sampling/weld station at deck level provide radiation protection for the facility worker (see Section A4.4.9) and interface with the MHM.

The sampling/weld station shielding consists of the center shield plate and two shield halves (Figure A2-40). The sampling/weld station shielding forms an inverted cone with a removable center shield plate. The center shield plate and shield halves are fabricated from 6-in.-thick carbon steel plate conforming to ASTM A36/A36M requirements. Before sampling or welding operations begin, the grating covers over the sampling/weld pit are removed and the sampling/weld station shielding is lowered into place by the sampling/weld station gantry crane. The shield halves are designed to provide closure of the sampling/weld pit, isolating it from the grade-level sampling/weld area concrete floor. The sampling/weld station shielding provides radiation protection for operating personnel. After sampling/weld station shielding is in place, the MHM aligns the MHM turret over the center shield plate, the center shield plate is retrieved into the MHM plug cavity, and the turret rotates and lowers the MCO into the sampling/weld station pit with the MCO supported by the sampling/weld station impact absorber inside the rotating shield. The center shield plate is then replaced by the MHM; the MHM moves away from the sampling/weld station; and the sampling/weld station shielding is removed by the sampling/weld station gantry crane to expose the top of the MCO for sampling and welding activities.

A sampling/weld station impact absorber (see Figure A2-40) consists of the following components: the top plate, the base plate, and energy-absorbing tube components in the middle. The top plate is 1.0-in.-thick carbon steel material conforming to ASTM A36/A36M. The base plate is constructed of a 0.5-in. carbon steel plate conforming to ASTM A36/A36M. Carbon steel material conforming to ASTM A513 is selected for the tube sections. The maximum diameter of the sampling/weld station impact absorber is 25.5 in. The overall height is about 18 in. after factory precrush of the assembled sampling/weld station impact absorber. The top and base plates serve to maintain the impact absorber centered in the sampling/weld station rotating shield. Galvanized steel cables spaced equidistantly around the absorber circumference maintain the tube components in place. Shear pins (12 total) at the rotating shield section below the MCO's canister collar will shear when impacted by an accidentally dropped MCO and allow the dropped MCO to impart the load to the absorber and rotating shield support structure, thus limiting the impact loads on the dropped MCO.
A4.4.20.3 Functional Requirements. The shield halves form an inverted cone with 75° angle of incline to the horizontal sampling/weld area surface. This angle effectively deflects an MCO that has been inadvertently, eccentrically dropped by the MHM into the sampling/weld rotating shield. The force of a dropped MCO is cushioned by the sampling/weld station impact absorber. Significant damage to the dropped MCO is prevented by compressing the impact absorber.

A sampling/weld station impact absorber must absorb the energy from a dropped MCO to the extent that deceleration forces to the MCO are below the specified design limits. The MCO’s DBA loading limits are provided in the MCO Topical Report (HNF-SD-SNF-SARR-005). The MCO is required to survive deceleration loads not to exceed 35 g. Without an impact absorber in the sampling/weld station pit, the dropped MCO is expected to exceed the 35 g specification limits of the MCO, similar to the cask receiving pit dropped MCO event (HNF-SD-SNF-DP-007, Appendix O). The sampling/weld station impact absorber is specified to limit the forces to the MCO from the maximum drop height above the operating deck to less than 35 g. The shear ring screws are required to shear and allow the dropped MCO to engage the impact absorber yet hold the MCO in place.

A4.4.20.4 System Evaluation. With the shield halves in place and the center shield plate removed by the MHM, the inverted cone shape of the opening into the rotating shielding mitigates the consequences of an accidentally dropped MCO. The thickness of the steel assemblies and the cone angle provide assurance that a dropped MCO will be guided into the rotating shielding cavity and cushioned by the sampling/weld station impact absorber.

The functional requirements listed above are provided in performance specifications to the supplier. The specified loaded MCO weight is 20,200 lb; the loaded MCO can be dropped from a maximum height of 20 ft, 3 in. without exceeding the 35 g specification limit or 679,000 lb force. The specification requirements include conformance with ANSI/AISC N690-94 for design and fabrication. Results of prototypical testing will demonstrate compliance with the functional requirements.

Analyses performed for the shear ring collar behavior during MCO drop conditions are documented in CSB-RM-0016. The results indicate that the dropped MCO energy will shear the screws at the scored reduced thickness points yet support the MCO in the sampling/weld station. The shear screws are furnished in accordance with ASTM F835 with a cut groove to 0.312 in. diameter, and provide a factor of safety of 2.7 to ensure that the ring does not break away during MCO placement.

A4.4.20.5 Controls (Technical Safety Requirements). The shield halves and sampling/weld station impact absorber are design features, and their safety function is managed by the configuration control process (see Section A5.6).
The following assumption associated with the sampling/weld station impact absorber requires TSRs to ensure performance of the safety functions.

- The sampling/weld station impact absorber provides protection of the MCO internals should an accidental drop of the MCO occur during lowering or raising of an MCO by the MHM. The presence of the sampling/weld station impact absorber is confirmed as part of initial construction acceptance. The sampling/weld station impact absorber needs verification of being in place before facility startup (see Sections A5.5.3.3 and A5.5.3.4).

- The shield halves need to be verified as being in place before the MHM lowers an MCO into the sampling/weld station pit for sampling or welding operations (see Section A5.5.3.3).

A4.4.21 Shield Hatch and Multi-Canister Overpack Guide Assembly

A4.4.21.1 Safety Function. The safety function of the shield hatch and MCO guide assembly is to deflect the MCO back into the cask receiving pit in the unlikely event the MHM drops the MCO onto the edge of the cask receiving pit. The shield hatch and MCO guide assembly protect the MCO from significant damage and provide radiation shielding for workers during cask lid removal and MCO removal operations.

A4.4.21.2 System Description. The shield hatch and MCO guide assembly unit consists of three parts: the shield hatch plate, the shield hatch ring, and the MCO guide assembly (Figure A2-17). The shield hatch and MCO guide assembly form an inverted cone plug with a removable shield plate. The shield hatch and MCO guide assembly are 66.25 in. in diameter. The MCO guide (fabricated of 8-in.-thick carbon steel) and the shield hatch ring and shield hatch plate (fabricated of 10-in.-thick carbon steel plate) conform to ASTM A36/A36M requirements. The MCO guide assembly has two drilled holes for alignment with the cask lid guide pins. The shield hatch and MCO guide assembly provide closure of the cask receiving pit, isolating it from the grade-level load-in/load-out area. The cask receiving pit is shown on Figure A2-9.

The tent gantry hoist is used to place the MCO guide assembly on the transportation cask after removal of the cask lid. The two centering holes of the MCO guide assembly align the MCO guide assembly over the center of the transportation cask. Next the shield hatch ring and shield hatch plate are lowered to floor level using the tent gantry hoist. This operation takes place after a transportation cask has been lowered into the cask receiving pit by the receiving crane and after the transportation cask lid has been removed. The shield hatch ring provides an interfacing surface for the MHM. The shield hatch ring has been designed to form a seal with the MHM's retractable nose. The shield hatch plate is designed to be removed by the MHM plug grapple. MHM interlocks prevent lifting the MCO unless the MHM is properly positioned over the shield hatch plate. The MHM's retractable nose in contact with the troweled concrete surface of the cask receiving pit provides shielding continuity during MCO retrieval operations.
A4.4.21.3 Functional Requirements. The shield hatch ring and MCO guide assembly form an inverted cone with a 75° and 71°, respectively, angle of incline to the horizontal load-in/load-out area surface. This angle effectively deflects an MCO, which has been inadvertently, eccentrically dropped by the MHM, back into the transportation cask. Significant damage to the dropped MCO is prevented by the MCO striking the bottom of the transportation cask with energy absorption by the cask and the cask receiving impact absorber.

The shield hatch and MCO guide assembly also perform a radiation shielding function for workers by sealing with the MHM’s retractable nose. The shielding goal for the cask receiving pit and the shield hatch and MCO guide assembly is to meet the ALARA principles of 10 CFR 835. The cask receiving pit and shield hatch and MCO guide assembly meet the requirements for personnel shielding safety provided in HNF-SD-SNF-DB-003 (Items 20 and 23).

A4.4.21.4 System Evaluation. With the shield hatch and MCO guide assembly in place and the shield hatch plate removed by the MHM, the inverted cone shape of the opening into the cask mitigates the consequences of an accidentally dropped MCO. The thickness of the steel assemblies and the cone angle provide assurance that a dropped MCO will be guided into the transportation cask cavity.

The shielding function of the shield hatch and MCO guide assembly ensures a safe area for operators to conduct cask receiving and MCO removal activities. Calculation CSB SH-2002, Floor Plug/Deck Interface Analysis, concluded that with the 10-in.-thick shield plate in place, the dose rate to the facility operator is less than the 0.05 mrem/h value calculated for the operating floor deck over vault 1. The design thicknesses of the shield hatch and MCO guide assembly are adequate to protect the facility worker and complies with Items 20 and 23 of HNF-SD-SNF-DB-003.

A4.4.21.5 Controls (Technical Safety Requirements). The following assumption associated with the shield hatch and MCO guide assembly requires TSRs to ensure performance of the safety functions.

- The shield hatch and MCO guide assembly are in place before the MCO is removed by the MHM. This is an MHM interface requirement, a dropped MCO mitigating requirement, and a worker protection requirement (Section A5.5.3.3).

A4.5 REFERENCES


ACI 301, 1996, *Specification for Structural Concrete*, American Concrete Institute, Detroit, Michigan.


ANSI/ACI 349-85, 1985, *Code Requirements for Nuclear Safety Related Concrete Structures*, American Concrete Institute, Detroit, Michigan.


Annex A — Canister Storage Building


Annex A — Canister Storage Building


Annex A — Canister Storage Building


CMAA 70-94, 1994, Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes, Crane Manufacturers Association of America, Charlotte, North Carolina.

Annex A — Canister Storage Building


Annex A — Canister Storage Building


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March 2000
HNF-3553 REV 0
Annex A — Canister Storage Building


HNF-3553 REV 0
Annex A — Canister Storage Building


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Figure A4-1. Design Basis Earthquake Response Spectra for Safety Class Items.

DBE RESPONSE SPECTRA FOR SAFETY CLASS ITEMS
Figure A4-2. Air Temperature Profile.
Figure A4-3. Operating Area, Storage Tube, and Vault Basemat Temperatures Expected from the Below-Grade Interim Storage of Multi-Canister Overpacks.

Operating Area

- \( T_{\text{Air}} = 85^\circ \text{F} \)
- \( T_{\text{Material Surface}} = 96^\circ \text{F} \)
- \( T_{\text{Concrete}} = 105^\circ \text{F} \)
- \( T_{\text{Plug}} = 129^\circ \text{F} \)
- \( T_{\text{Concrete}} = 121^\circ \text{F} \)
- \( T_{\text{Tube}} = 155^\circ \text{F} \)
- \( T_{\text{Air}} = 139^\circ \text{F} \) (HOT SPOT)

Storage Tube

- \( T_{\text{MCO Surface}} = 230^\circ \text{F} \)
- \( T_{\text{Tube}} = 186^\circ \text{F} \)

Impact Absorbers

- T Contact \( \leq 130^\circ \text{F} \)
- Tube Base

OPERATING AREA, STORAGE TUBE AND VAULT BASEMAT TEMPERATURES
Figure A4-4. Design Basis Earthquake Response Spectra Curves for the Multi-Canister Overpack Handling Machine Crane.
Figure A4-5. Multi-Canister Overpack Handling Machine Power Distribution.
(sheet 1 of 2)
Figure A4-5. Multi-Canister Overpack Handling Machine Power Distribution.
(sheet 2 of 2)
Figure A4-6. Schematic of Multi-Canister Overpack Handling Machine Control and Protection System.

**CONTROL SYSTEM LOGIC**

I) Interlock Channel Y

LIMIT SWITCHES, RESOLVERS, ETC

Y CONTACTOR

COMMANDS

LIMIT SWITCHES, RESOLVERS, ETC

CONTROL SYSTEM LOGIC

II) Interlock Channel X

X CONTACTOR

MOTOR, ETC

"N" CLASS PROTECTION = X + Y CHANNELS
"P" CLASS PROTECTION = X CHANNEL
Figure A4-7. Design Basis Earthquake Response Spectra Curves for the Receiving Crane. (sheet 1 of 3)

Response Spectrum for the Receiving Crane
North-South Direction

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Note: This Spectrum is based on 3D-SSI Analysis (CSB-S-0052) using "SASSI" computer code and ISRS developed in calculation CSB-S-0053
Response Spectrum for the Receiving Crane
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Note: This Spectrum is based on 3D-SSI Analysis (CSB-S-0052) using "SASSI" computer code and ISRS developed in calculation CSB-S-0053

Figure A4-7. Design Basis Earthquake Response Spectra Curves for the Receiving Crane. (sheet 3 of 3)
Figure A4-8. Design Basis Earthquake Response Spectra for the Multi-Canister Overpack Sampling/Weld Station Gantry Crane. (sheet 1 of 3)
Figure A4-8. Design Basis Earthquake Response Spectra for the Multi-Canister Overpack Sampling/Weld Station Gantry Crane. (sheet 2 of 3)

**MCO SAMPLING/WELD STATION GANTRY CRANE DESIGN**

**Response Spectra**

(5% Damping)

**HORIZONTAL EAST-WEST DIRECTION**

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AF4-12  March 2000
Figure A4-8. Design Basis Earthquake Response Spectra for the Multi-Canister Overpack Sampling/Weld Station Gantry Crane. (sheet 3 of 3)
Figure A4-9. Design Basis Earthquake Response Spectra for Safety Significant Items.
Figure A4-10. Multi-Canister Overpack Drop Sequence.
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Section 3 of 3

### Document Information

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CHAPTER A5.0

DERIVATION OF TECHNICAL SAFETY REQUIREMENTS
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<td>Administrative Control</td>
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<tr>
<td>ALARA</td>
<td>as low as reasonably achievable</td>
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<td>BED</td>
<td>Building Emergency Director</td>
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<td>CSB</td>
<td>Canister Storage Building</td>
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<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
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<td>FSAR</td>
<td>final safety analysis report</td>
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<td>LCO</td>
<td>Limiting Condition for Operation</td>
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<td>MCO</td>
<td>multi-canister overpack</td>
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<td>MHM</td>
<td>multi-canister overpack handling machine</td>
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<td>NPH</td>
<td>natural phenomena hazard</td>
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<td>SNF</td>
<td>spent nuclear fuel</td>
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<tr>
<td>SSC</td>
<td>structure, system, and component</td>
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<td>TSR</td>
<td>Technical Safety Requirement</td>
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A5.0 DERIVATION OF TECHNICAL SAFETY REQUIREMENTS

A5.1 INTRODUCTION

A description of the essential features of Spent Nuclear Fuel (SNF) Project derivation of Technical Safety Requirements (TSRs) is provided in Section 5.1 of the SNF Project Final Safety Analysis Report (FSAR).

A5.2 REQUIREMENTS

The requirements that form the basis for the SNF Project derivation of TSRs are identified in Section 5.2 of the SNF Project FSAR.

A5.3 TECHNICAL SAFETY REQUIREMENTS COVERAGE

The suite of TSRs for analyzed hazards and accidents is summarized in Table A5-1. This table lists TSR controls in accordance with the accident analyses in Chapter A3.0. Table A5-1 is written to provide a road map from the respective accident analysis section to the relevant subheadings within Section A5.5, where derivation details are arranged by TSR control.

The necessary and sufficient TSR controls are established based upon consideration for public safety, significant defense in depth, significant worker safety, and for maintaining radiological consequences below release limits and risk evaluation guidelines. Section 5.3 of the SNF Project FSAR contains details applicable to all SNF Project facilities. Section A5.3.2 contains information specific to the Canister Storage Building (CSB) in addition to that provided in Section 5.3.2.

A5.3.1 Criteria

The control selection criteria used for the SNF Project are described in Section 5.3.1 of the SNF Project FSAR.
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Table A5-1. Hazard and Accident Analyses and Technical Safety Requirement Cross Reference. (2 sheets)

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AC = Administrative Control.
CSB = Canister Storage Building.
FSAR = Safety analysis report.
LCO = Limiting Condition for Operation.
MCO = Multi-canister overpack.
MHM = Multi-canister overpack handling machine.
A5.3.2 Safety Structures, Systems, and Components not Provided
with Technical Safety Requirement Coverage

The safety-class and safety-significant structures, systems, and components (SSCs) that
have not been provided with TSR coverage are identified below. All other safety SSCs are
provided with TSR coverage with a Limiting Condition for Operation (LCO) or are included in an
Administrative Control (AC) program. The bases for not providing TSR coverage are provided
in the following paragraphs.

A5.3.2.1 Transportation Cask. The transportation cask is a safety-class SSC. There are no
TSR controls for this safety SSC because it is a Design Feature (refer to Section A5.6, “Design
Features”) and its safety function is managed by the configuration control process. In
Chapter A3.0 this SSC was classified as providing significant defense in depth by maintaining its
structural integrity.

A5.3.2.2 Multi-Canister Overpack. The multi-canister overpack (MCO) is a safety-class SSC.
There are no TSR controls for this safety SSC because it is a Design Feature (refer to
Section A5.6) and its safety function is managed by the configuration control process. In
Chapter A3.0 this SSC was classified as providing significant defense in depth by maintaining its
structural integrity. It also was credited in Chapter A6.0 as having certain geometrically favorable
design characteristics.

A5.3.2.3 Canister Storage Building Subsurface Structures. The CSB subsurface structures
(vault 1, air intake and exhaust plenums) are classified as safety-class SSCs. There are no TSR
controls for these safety SSCs because they are Design Features (refer to Section A5.6) and their
safety functions are managed by the configuration control process. These SSCs were identified as
maintaining structural integrity throughout all design basis natural phenomena hazard (NPH)
events.

A5.3.2.4 Carbon Steel Basemat Embeds. The carbon steel basemat embeds are classified as
safety-class SSCs. There are no TSR controls for these safety SSCs because they are Design
Features (refer to Section A5.6) and their safety functions are managed by the configuration
control process. These SSCs were identified as maintaining structural integrity throughout all
design basis NPH events.

A5.3.2.5 Canister Storage Building At-Grade Structures. The operating area deck,
sampling/weld area, load-in/load-out area, intake structure, and exhaust stack are classified as
safety-class SSCs. The operating area shelter, support area foundation, cask receiving pit, and
sampling/weld station MCO support structure are classified as safety-significant SSCs. There are
no TSR controls for these safety SSCs because they are Design Features (refer to Section A5.6)
and their safety functions are managed by the configuration control process. The safety-class
SSCs were identified as maintaining structural integrity throughout all design basis NPH events,
and the safety-significant SSCs were identified as not failing in such a way as to cause the failure
of any safety-class SSC.
A5.3.2.6 Standard Storage Tubes, Lower Flanges, Tube Base Assemblies, and Tube Plugs. The standard storage tubes and tube base assemblies are classified as safety-class SSCs, and the lower flanges and tube plugs are classified as safety-significant SSCs. There are no TSR controls for these safety SSCs because they are Design Features (refer to Section A5.6) and their safety functions are managed by the configuration control process. These SSCs were identified as maintaining structural integrity throughout all design basis NPH events.

A5.3.2.7 Overpack Storage Tubes, Lower Flanges, Tube Base Assemblies, and Overpack Tube Plugs. The overpack storage tubes and tube base assemblies are classified as safety-class SSCs, and the lower flanges and overpack tube plugs are classified as safety-significant SSCs. There are no TSR controls for these SSCs because they are Design Features (refer to Section A5.6) and their safety functions are related to recovery actions, not to accidents during normal operations. Controls related to specific recovery actions must be developed on a case-by-case basis depending on current conditions and recovery evaluation results.

A5.3.2.8 Helium Rupture Disk. The helium rupture disk PSE-1 is classified as a safety-significant SSC. There are no TSR controls for this SSC because it is a Design Feature (refer to Section A5.6) and its safety function is managed by the configuration control process. This SSC was identified as relieving pressures at the sampling/weld station.

A5.3.2.9 Receiving Crane Structure and Hoist. The receiving crane structure and hoist are classified as safety-significant SSCs. There are no TSR controls for these safety SSCs because they are Design Features (refer to Section A5.6) and their safety functions are related to recovery actions, not to accidents during normal operations. Controls related to specific recovery actions must be developed on a case-by-case basis depending on current conditions and recovery evaluation results.

A5.3.2.10 Multi-Canister Overpack Handling Machine Structural Components and Multi-Canister Overpack Hoist and Grapple. The MCO handling machine (MHM) structural components and MCO hoist and grapple are classified as safety-significant SSCs. There are no TSR controls for these safety SSCs because they are Design Features (refer to Section A5.6) and their safety functions are managed by the configuration control process. These SSCs are credited in the accident analyses because of their contribution to reducing drop frequencies.

A5.3.2.11 Shielding Features of the Transportation Cask, Multi-Canister Overpack Shield Plug, Multi-Canister Overpack Handling Machine, and Operating Deck. The shielding features of the transportation cask, MCO shield plug, MHM, and operating deck that protect workers from direct radiation from the MCO contents are classified as safety-significant. There are no TSR controls for these safety SSCs because they are fixed Design Features (refer to Section A5.6) and their safety functions are managed by the configuration control process. These SSCs were identified as providing significant worker safety in Chapter A3.0.

A5.3.2.12 Tube Vent and Purge Cart. The tube vent and purge cart is used in support of recovery actions, and the pressure boundary in the sampling path is classified as safety significant in this role. There are no TSR controls for this SSC because its safety function is related to
recovery actions, not to accidents during normal operations. Controls related to specific recovery actions must be developed on a case-by-case basis depending on current conditions and recovery evaluation results.

A5.3.2.13 Transportation Cask Servicing System. The transportation cask servicing system is used in support of recovery actions and the flex connector, high-efficiency particulate air filter, piping between the filter and the cask, and pressure safety valve PSV-102 are classified as safety-significant in this role. There are no TSR controls for this SSC because its safety function is related to recovery actions, not to accidents during normal operations. Controls related to specific recovery actions must be developed on a case-by-case basis depending on current conditions and recovery evaluation results.

A5.4 DERIVATION OF FACILITY MODES

A5.4.1 Operational Modes

The facility operational modes (Operation and Standby) for the CSB were identified to indicate overall facility status so that conditions can be clearly highlighted to CSB personnel.

The facility modes for the CSB are defined as follows.

Operation — The facility is capable of receiving MCOs or transferring MCOs between locations within the facility. Routine operational and maintenance activities may be performed (e.g., surveillances, equipment checks). Major planned maintenance activities (e.g., work requiring the replacement or temporary disassembly of major components such as MHM interlock controls) are not allowed. Corrective maintenance to restore operability as part of an LCO-required action is allowed. Any other operational evolutions (e.g., sampling, welding) already in progress must be completed, or terminated in accordance with procedures to return to a stable condition, before moving to Standby mode.

Standby — The facility contains MCOs. Receipt of new MCOs, transfers of MCOs between locations within the facility, and MCO welding or sampling activities are not allowed, except as part of an approved recovery plan. (Note: The facility may not enter the Standby mode while an MCO is located within the MHM.) Routine operational and maintenance activities may be performed (e.g., surveillances, equipment checks). Major maintenance activities (e.g., work requiring the replacement or temporary disassembly of major components such as MHM interlock controls) may be performed.

The facility may be manned or unmanned in either mode. To be unmanned, all MCOs within the facility must be within a standard storage tube, the sampling/weld station, or the cask receiving pit. When the CSB is unmanned, MCOs are not allowed to be attached to CSB lifting
devices, and welding or sampling activities are not allowed to be in progress (i.e., they must have been completed or terminated).

A5.4.2 Minimum Staffing Levels

When the CSB is manned, the minimum operations shift complement for both the Operation and Standby modes is one shift manager (or designee) and two nuclear operators. In Standby mode, the required minimum CSB staff can be composed of any SNF Project shift managers or nuclear operators who have had the appropriate CSB-specific training, regardless of whether those staff members are physically located at the CSB or at another SNF Project facility. This minimum staff in each mode is considered adequate to perform the minimum safety functions necessary to protect the health and safety of the public, onsite workers, and the environment during normal operations, and abnormal and emergency conditions. The shift manager (or designee) is required to be Building Emergency Director (BED)-qualified.

Qualification training for the minimum staff (managers and nuclear operators) is addressed in Chapter 12.0, “Procedures and Training,” of the SNF Project FSAR. The program for TSR, emergency, and alarm response administrative procedures also is addressed in Chapter 12.0 of the SNF Project FSAR. Emergency response is addressed in Chapter 15.0, “Emergency Preparedness Program,” of the SNF Project FSAR. The general considerations for determining the minimum staff are provided in Section 5.4.2 of the SNF Project FSAR, while CSB-specific considerations are identified below.

Normal operations. Staffing requirements for the minimum operations shift complement during normal operations is based on accomplishment of the routine surveillances required for TSR compliance. A surveillance is considered a low-difficulty task. The surveillances that must be performed during normal operations include the following:

- Verifications that systems are operable
- Verifications that equipment is in place prior to beginning specific operational tasks
- Instrument calibrations
- Functional tests of systems to demonstrate operability.

The minimum operations shift complement is adequate during normal operations to perform necessary job functions.

Abnormal conditions. The minimum operations shift complement during abnormal conditions is necessary to ensure compliance with required actions specified in LCO action statements with completion times of less than 8 hours or that include the term “immediately.” Hanford Site experience has shown that additional staff could be provided within 8 hours, if needed (considering the most adverse weather and travel conditions), to ensure all LCO completion times are met. LCO completion times of “immediately” imply the highest sense of
urgency and are given top priority over all other activities. The minimum staff is not given tasks that could interfere with meeting TSR requirements.

The minimum operations shift complement is the same in either Operation or Standby mode. Specific LCO-required actions that would need to be performed immediately within a specified completion time, or prior to resuming operation (in this case a completion time is not specified), include the following:

- Shutdown of MHM operation
- Verify operating conditions (e.g., change of component status such as valve closure)
- Restore system operability.

The minimum operations shift complement is adequate during abnormal conditions to perform necessary job functions.

Emergency conditions. The minimum operations shift complement during emergency conditions is necessary to ensure response to the spectrum of accidents analyzed in Chapter A3.0 (hazardous and nonhazardous). The minimum operations shift complement must make prompt initial notifications and implement initial protective actions to preclude or reduce the exposure of individuals affected by hazards or unsafe conditions during an emergency.

The shift manager (or designee) is the BED. The BED is primarily responsible for assessment of the event and protective actions at the CSB. The BED also makes onsite notifications, implements emergency management procedures, implements facility emergency plans, classifies events, and controls event response. Two nuclear operators are considered adequate to respond to an event scene and support actions requested by the BED. The BED requests support services, as necessary, to perform administrative functions and the minimum functions required to ensure the health and safety of the public, onsite workers, and the environment.

The minimum operations shift complement is adequate to meet applicable emergency preparedness requirements.

A5.5 TECHNICAL SAFETY REQUIREMENT DERIVATION

A5.5.1 Multi-Canister Overpack Handling Machine

A5.5.1.1 LCO 3.1.1 — Multi-Canister Overpack Handling Machine Shear and Drop Prevention Interlock System.

Purpose. This control, to ensure MHM shear and drop prevention interlock system operability, is derived from the mechanical damage of MCO accident in Section A3.4.2.1.
Shearing the MCO or dropping it from an excessive height potentially results in a loss of SNF confinement. The P2 interlock is relied on as a significant defense-in-depth feature to limit the susceptibility of an MCO to breach caused by inadvertent MHM movement and as significant worker safety protection to limit personnel exposure to direct radiation from MCOs stored in the storage tubes. In addition, the P6 and P21 interlocks provide safety-significant defense-in-depth protection against shears caused by seismic events. The remaining interlocks in this system that are identified in Chapter A3.0 (P3, P5, P8, P9, P26, P57, P61, P62, P63, P65, P66, P80, and P85) provide additional defense in depth (non-TSR) for their role in contributing to the low likelihood of drops and shears throughout the facility. A summary of the interlocks addressed by this LCO follows:

- The P2 interlock to ensure that the storage tube plug has been placed in a storage tube before allowing shield skirt raising and bridge and trolley travel.
- The P6 interlock to ensure that the limit switch is not actuated and the resolver and programmable limit switch do not indicate a fully raised position unless the top of the MCO grapple assembly is less than the required distance below the outside bottom of the MCO hoist and drive assembly box.
- The P21 interlock to ensure that the switches are not actuated unless the right hand face of the trolley seismic restraint pin is within its acceptance distance below the lower face of the mounting plate and the bridge seismic clamps are fully disengaged.

This LCO applies in the Operation mode. The LCO requires operability of each interlock, including sensors, relays, and contactors, that is identified in Chapter A3.0 to reduce the likelihood of MCO shear. Operability requirements include the following:

- Prior to MHM operation after startup, a channel check of the P6 and P21 interlocks will be performed to give assurance that the MHM-supported subsystem equipment functions as intended.
- A quarterly channel test will be performed to demonstrate the proper response of the interlock to an injected signal (simulated or actual) to verify the correct channel response.
- Annual calibration and functional test of each interlock to ensure that all required inputs and their expected responses are adequate to verify operability.

**Derivation Criteria.** These controls were selected because they prevent, rather than mitigate, the associated accidents and use engineered controls rather than administrative programs to achieve their safety functions.
A5.5.2 Confinement Systems

A5.5.2.1 LCO 3.2.1 — Sampling Hood Exhaust System.

**Purpose.** This control, to ensure sampling hood exhaust system operability, is derived from the MCO external hydrogen deflagration accident in Section A3.4.2.4. The release of radioactive material without controls is below risk evaluation guidelines. However, the worker safety considerations of an external deflagration warrant the elevation of the sampling hood exhaust system to safety significant. Ensuring the operability of this system reduces the inherent risk to worker safety.

The sample hood confinement system controls consist of operability requirements that include:

- Instrumentation necessary to determine that sample hood flow rates are adequate
- Integrity of the ducting up to and including exhaust fans.

This LCO applies in the Operation mode when the MCO process port valve is open to the sampling/weld station equipment. Surveillance requirements associated with this LCO include the following:

- Calibration of the flow instrumentation as established by the SNF Project Preventive Maintenance/Testing Program
- An operability inspection before MCO sampling operations are begun — This inspection includes sample hood glove ports for degradation, flexible hose and connections for integrity, proper balance damper and isolation damper positioning. Part of the operability inspection includes verification of a 100 ft³/min flow rate through the sample hood prior to MCO sampling. This verification ensures that the hood exhaust system flow is operating before opening the MCO process port. Although the safety analyses in Chapter A3.0 show that flow on the order of 5 ft³/min is sufficient, 100 ft³/min was chosen as the minimum value because it represents the minimum design flow value. Flow rates below this value are indicative of problems with the hood exhaust and bring into question the reliability of the system to perform its safety function throughout the sampling operations.

**Derivation Criteria.** This control was selected based on significant worker safety considerations in Chapter A3.0.

A5.5.2.2 LCO 3.2.2 — Sampling Piping Confinement System.

**Purpose.** This control, to ensure sampling piping confinement system operability, is derived from the MCO external hydrogen deflagration accident in Section A3.4.2.4. The release
of radioactive material without controls is below risk evaluation guidelines. However, the worker safety considerations of an external deflagration warrant the elevation of the sampling piping confinement system to safety significant. Ensuring the operability of this system reduces the inherent risk to worker safety.

The sampling piping confinement system controls consist of operability requirements to ensure the integrity of the sampling piping. Surveillance requirements associated with this LCO include the following:

- A pressure check of the sampling piping and MCO connection to verify that the system leakage rate is less than 40 cm³/s at 75 lb/in² gauge (as confirmed by a pressure decay rate) before taking an MCO sample.

This LCO applies in the Operation mode when the MCO process port valve is open to the sampling/weld station equipment.

**Derivation Criteria.** This control was selected based on significant worker safety considerations in Chapter A3.0.

### A5.5.3 Administrative Controls

AC 5.1 through AC 5.6 are addressed in Chapter 5.0 of the SNF Project FSAR.

### A5.5.3.1 AC 5.7 — Nuclear Criticality Safety

**Purpose.** This control protects the assumptions of the nuclear criticality evaluation in Chapter A6.0 to ensure that CSB operations minimize the risk of nuclear criticality. The purpose of this AC is to protect double contingency features that are relied upon to preclude a criticality event at the CSB. Key elements of this programmatic AC are derived from contractor procedures that include requirements for criticality safety evaluations, criticality prevention specifications, pre-fire plans, and criticality training. The criticality controls identified in Chapter A6.0 are derived from HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*.

**Derivation Criterion.** A nuclear criticality program is required according to DOE Order 5480.22, *Technical Safety Requirements*, Section 9.e.(5).

### A5.5.3.2 AC 5.8 — Measurement and Test Equipment

**Purpose.** This control provides a support control for the measuring and test equipment used to verify process parameters to comply with the TSRs when specific instrumentation and associated surveillances are not included in the TSR. This control applies to (1) optional instrument systems allowable in accordance with the LCO bases, and (2) instrumentation used to
verify parameters for the ACs in the TSR. This control on the measuring and test equipment ensures that process parameters are properly monitored.

The key elements of the program include maintaining identification and traceability of TSR-related instrumentation and equipment, periodic testing of the instrumentation and equipment used, and maintaining records that demonstrate that the instrumentation and equipment were used while in calibration.

The program applies to installed and portable measuring and test equipment when used to verify parameters specified in the TSRs where specific instrumentation and associated surveillances are not already identified in the TSR. This program does not apply to equipment such as rulers, tape measures, levels, and other measurement devices if commercial equipment provides accuracy adequate to meet the measurement specifications.

**Derivation Criteria.** This control was included because for some parameters in the TSR, it is not essential to specify the particular instrumentation or measuring equipment used to demonstrate compliance with the parameter involved provided the instrumentation or equipment used to monitor the parameter is maintained. This support control was selected because it provides operational flexibility to use various instrumentation and equipment to measure TSR parameters provided it complies with this AC while ensuring the safety functions assumed in the accident analysis are met.

**A5.5.3.3 AC 5.9 — Canister Storage Building Operational Controls.**

**Purpose.** This AC focuses on the importance of those planned operational steps that were identified as critical assumptions in the accident analyses in Chapter A3.0 or that provided significant defense in depth. The key elements of the AC include the following.

- Requirement to use the proper yoke (the cask lifting yoke) when moving the cask-MCO with the receiving crane — This provides significant defense in depth by protecting the lift height of the bottom of the cask-MCO to a maximum of 60 in. above the load-in/load-out area floor.

- Visual verification that the MHM seismic restraints (bridge, trolley, and turret) are applied before MCO raising or lowering operations begin — This ensures that the MHM is capable of performing properly during a seismic event to provide significant defense in depth to reduce the likelihood of MCO shearing as well as to maintain initial condition assumptions used to demonstrate that the MHM is not susceptible to catastrophic failure.

- Verification that rail frog hold-down bolts are snug when the rail frog is repositioned — This ensures that the rail frog is capable of performing properly during a seismic event to provide significant defense in depth to reduce the likelihood of MCO
shearing as well as to maintain initial condition assumptions used to demonstrate that
the MHM is not susceptible to catastrophic failure.

- Surveillance of the rail clip hold-down bolts for relaxation in accordance with
  recommendations from the designer — Maintenance ensures the proper configuration
  of the rail clips as a structural Design Feature to provide significant defense in depth
to reduce the likelihood of MCO shearing as well as to maintain initial condition
assumptions used to demonstrate that the MHM is not susceptible to catastrophic
failure.

- Verification that the sampling/weld station shield halves are in place before placing an
MCO in or removing an MCO from the sampling/weld station — This ensures the
proper configuration of the shielding as a Design Feature to provide significant
defense-in-depth protection against eccentric MCO drops and to provide worker
safety from direct radiation doses during MCO placement and removal.

- Verification that the interface guide ring funnel is in place before placing an MCO in
or removing an MCO from a storage tube — This ensures the proper configuration
of the interface guide ring funnel as defense in depth against eccentric MCO drops.

- Verification that the shield hatch and MCO guide assembly is in place before the
MCO is removed from the cask receiving pit by the MHM — This ensures the proper
configuration of the shielding as a Design Feature to provide significant defense-in-
depth protection against eccentric MCO drops and to provide worker safety from
direct radiation doses during MCO removal.

- Verification that the MCO centering guide is properly installed in the MHM before
the MHM leaves the maintenance pit in the MCO mode — This ensures the proper
configuration of the MCO centering guide as a Design Feature to provide significant
defense-in-depth protection against eccentric MCO drops.

This AC heightens attention to the need for CSB compliance with these planned operational steps.

Derivation Criterion. These controls are necessary to support the assumptions made
within the safety analysis. These assumptions rely upon operator actions to ensure that they are
valid.

A5.5.3.4 AC 5.10 — Impact Absorber Availability.

Purpose. The impact absorber availability control is derived from the accident analyses for
mechanical damage of MCO in Section A3.4.2.1. The impact absorbers in the cask receiving pit,
storage tube, and sampling/weld station play a significant defense-in-depth role to prevent damage
to MCOs handled at the CSB.
This control requires that a program be established and maintained to ensure that the following key elements occur.

- Verify that impact absorbers are installed in the sampling/weld station, cask receiving pit, and storage tubes (bottom impact absorbers only) before facility startup.
- Verify that intermediate impact absorbers are placed in any storage tube that is to be used to store a second MCO.
- Establish procedures to ensure prompt replacement of damaged impact absorbers.

**Derivation Criterion.** This control is necessary to support the safety analysis regarding mitigation of MCO drops. Placement of impact absorbers within these locations is an administrative feature and best handled through an AC.

**A5.5.3.5 AC 5.11 — Storage Tube Plug Placement.**

**Purpose.** This control is derived from the evaluation of significant worker safety in Section A3.3.2.3.2 of Chapter A3.0. Direct radiation doses from the MCO are significant and shielding is required at all times. This shielding is provided by the tube plugs during MCO storage.

This control requires that a program be established and maintained to ensure that standard storage tube plugs and overpack storage tube plugs are properly placed at all times during facility operation, and are in good working condition. This involves verification of placement before facility startup and procedures to deal promptly with tube plugs that may need replacement. The only acceptable method of removal of the tube plugs is with the MHM, which provides shielding during this operation. Replacement of a tube plug is necessary before the MHM moves from the tube location affected.

**Derivation Criterion.** This control was selected to protect the facility worker from significant direct radiation exposure during operations. Proper placement of all tube plugs at the start of operations is assumed within the safety analysis report to protect facility workers from shine effects from storage tubes that may be near the otherwise open storage tubes.

**A5.5.3.6 AC 5.12 — Receipt Acceptance Criteria.**

**Purpose.** This control protects the accident analysis assumptions in all the evaluated accidents. The safety analysis assumes that all MCOs received meet the minimum requirements for acceptance.

This AC requires a program to verify the acceptability of an MCO before receipt by CSB personnel. A key element of this program is verification before acceptance of each MCO delivery to ensure that the proper processing steps and conditions have been carried out for each MCO.
The program also must address recovery planning activities for out-of-specification MCOs that arrive at the CSB.

**Derivation Criteria.** This control was selected to protect major analysis assumptions (e.g., maximum particulate content, maximum leakage rate, maximum water content) that, if altered, could significantly affect consequences.

A5.5.3.7 AC 5.13 — Combustible Loading Limits.

**Purpose.** The combustible loading limits control is established based on Chapter A3.0 assumptions made regarding combustible loading limits determined in HNF-SD-SNF-FHA-002, *Final Fire Hazard Analysis for the Canister Storage Building*, and in its implementation plan.

This control applies in the Operation and Standby modes. A program shall be established and maintained to limit the combustible loadings as determined by the fire hazard analysis (HNF-SD-SNF-FHA-002) and implementation plan.

**Derivation Criteria.** This control was selected to protect Chapter A3.0 assumptions relating to fire hazards that, if altered, could significantly affect consequences.

A5.5.3.8 AC 5.14 — Configuration Management of Design Features.

**Purpose.** This configuration management AC was deemed necessary to protect the safe operation of the facility against uncontrolled changes to Design Features. A sitewide institutional safety program for configuration management is in place to control such changes. In addition, the SNF Project has developed its own plan, HNF-SD-SNF-CM-001, *Spent Nuclear Fuel Project Configuration Management Plan*, to handle configuration management at the SNF Project facilities. The plan defines the process that the SNF Project has approved to secure configuration management, and it is formatted to include the elements and functions established in DOE-STD-1073-93, *Guide for Operational Configuration Management Program*. The plan includes compliance requirements with the overall Site configuration management system and criteria for change control, document control, configuration management implementation, and periodic assessments. This AC heightens attention to the need for CSB compliance with these configuration management plans.

**Derivation Criteria.** This control was deemed necessary to protect the safe operation of the facility against uncontrolled changes to Design Features. HNF-PRO-700, *Safety Analysis and Technical Safety Requirements*, states that configuration management is a candidate AC when Design Features have been identified for a facility.
A5.6 DESIGN FEATURES

Design Features for the CSB that, if altered or modified, would have a significant effect on safe operation are listed below. Descriptions of these Design Features are provided in Chapter A2.0, "Facility Description." The safety functions they perform are provided in Chapter A4.0, "Safety Structures, Systems, and Components." Chapters A2.0 and A4.0 should be consulted for details.

A5.6.1 Transportation Cask

Design Features for the transportation cask include the following.

- The cask structure provides protection to the MCO from shearing forces while being lowered into the cask receiving pit.
- The cask shielding, in conjunction with the receiving crane shielded operator station, provides attenuation of gamma and neutron radiation to limit facility worker exposure to levels consistent with site administrative limits and ALARA (as low as reasonably achievable) practices.
- Under normal conditions, the cask is leaktight and capable of maintaining confinement as identified in Chapter A3.0.

A5.6.2 Multi-Canister Overpack

Design Features for the MCO include the following.

- The MCO shell, filter guard plate, shield plug, Mark IA fuel and scrap baskets' center posts, and baseplates provide a geometrically favorable SNF configuration in regard to criticality issues.
- Under normal conditions, the MCO is capable of maintaining confinement with a leakage rate of no greater than $10^{-4}$ cm$^3$/s, as required by HNF-SD-SNF-SARR-005, *Multi-Canister Overpack Topical Report*.
- The MCO shield plug shielding provides attenuation of gamma and neutron radiation to limit facility worker exposure to levels consistent with site administrative limits and ALARA practices.
A5.6.3 Canister Storage Building Subsurface Structures

Design Features for the CSB subsurface structures (vault 1, air intake and exhaust plenums) include the following.

- The structural features maintain structural integrity throughout all design basis NPH events.

A5.6.4 Carbon Steel Basemat Embeds

Design Features for the carbon steel basemat embeds include the following.

- The structural features maintain structural integrity throughout all design basis NPH events.

A5.6.5 Canister Storage Building At-Grade Structures

Design Features for the CSB at-grade structures (operating area deck, sampling/weld area, sampling/weld station MCO support structure, load-in/load-out area, cask receiving pit, support area building foundation, intake structure, exhaust stack, and operating area shelter) include the following.

- The structural features of the operating deck, sampling/weld area, load-in/load-out area, intake structure, and exhaust stack maintain structural integrity throughout all design basis NPH events.

- The operating area shelter, support area building foundation, sampling/weld station MCO support structure, and cask receiving pit will not fail during a design basis NPH event in such a way as to cause the failure of any safety-class SSC.

A5.6.6 Standard Storage Tubes, Lower Flanges, Tube Base Assemblies, and Tube Plugs

Design Features for the standard storage tubes, lower flanges, tube base assemblies, and tube plugs include the following.

- The structural features maintain structural integrity throughout all design basis NPH events.

- The lower flange of the storage tube has a 45° incline.
The tube plug geometry is such that the plug is incapable of passing through the lower flange and impacting an MCO.

A5.6.7 Overpack Storage Tubes, Lower Flanges, Tube Base Assemblies, and Overpack Tube Plugs

Design Features for the overpack storage tubes, lower flanges, tube base assemblies, and overpack tube plugs include the following.

- The structural features maintain structural integrity throughout all design basis NPH events.
- The lower flange of the storage tube has a 45° incline.

A5.6.8 Shielding Features of the Shield Hatch and Multi-Canister Overpack Guide Assembly, Multi-Canister Overpack Handling Machine, Operating Deck, Tube Plugs, and Sampling/Weld Station Shield Halves and Center Shield Plate

Design Features for the shield hatch and MCO guide assembly, MHM, operating deck, tube plugs, and sampling/weld station shield halves and center shield plate include the following.

- The shielding provides attenuation of gamma and neutron radiation to limit facility worker exposure to levels consistent with site administrative limits and ALARA practices.

A5.6.9 Receiving Crane Structure and Hoist

No individual Design Feature of the receiving crane is more important than another to protect its safety function. Therefore no individual features are listed here.

A5.6.10 Multi-Canister Overpack Handling Machine

Design Features for the MHM include the following.

- The MHM seismic restraints maintain structural integrity throughout a design basis seismic event.
- The MHM rails and rail frogs maintain structural integrity throughout a design basis seismic event.
• The MHM MCO grapple is designed to preclude releasing an MCO (disengaging) unless the load is no longer supported by the grapple.

A5.6.11 Sampling/Weld Station

Design Features of the sampling/weld station include the following.

• Pressure safety relief device PSE-1 is specified to relieve at 150 lb/in² gauge.
• The sampling/weld station shield halves have a 75° incline.
• The center shield plate geometry is designed such that the center shield plate is incapable of passing through the shield halves and impacting an MCO.

A5.6.12 Shield Hatch and Multi-Canister Overpack Guide Assembly

Design Features of the shield hatch and MCO guide assembly include the following.

• The shield hatch geometry is designed such that the shield hatch is incapable of passing through the MCO guide assembly and impacting an MCO.
• The shield hatch and MCO guide assembly have a 75° incline.

A5.6.13 Gas Cylinder Valves

Design Features of all gas cylinders valves include the following:

• A maximum hole size in the inlet to the butt of the cylinder valve limits the thrust produced by escaping gas to an amount that cannot accelerate the cylinders to dangerous velocities.

A5.6.14 Site Grading

Design Features for site grading include the following.

• Site grading is maintained such that the probable maximum precipitation event (9.2 in. in 6 hours) does not allow water to flow into the CSB operating area.
• Site grading is maintained such that the accumulation of runoff water cannot result in the flow of significant water into the CSB operating area.
A5.6.15 Cask Lifting Yoke

Design Features for the cask lifting yoke include the following.

- The length of the cask lifting yoke limits the lift height of the cask-MCO such that the receiving crane cannot lift the bottom of the cask more than 60 in. above the floor.

A5.7 INTERFACES WITH TECHNICAL SAFETY REQUIREMENTS FROM OTHER FACILITIES

The TSRs of other facilities that affect the SNF Project facilities safety basis are summarized in Section 5.7 of the SNF Project FSAR. TSRs of other facilities that affect the CSB are summarized in the following subsections.

A5.7.5 Receipt of Spent Nuclear Fuel

The CSB interfaces with other facilities through the receipt of SNF from the Cold Vacuum Drying Facility (CVDF). Chapter A3.0 identifies facility interfaces that are key to protecting the CSB safety basis. As discussed in Section 5.7.5.2 of the SNF Project FSAR, SNF shipments from the CVDF must meet the criteria and assumptions identified in Chapter A3.0.

The CVDF sends SNF to the CSB based upon dryness testing. HNF-3673, *Cold Vacuum Drying Facility Technical Safety Requirements*, requires that proper testing be conducted to establish that the MCO water content is within acceptable levels to ensure that key parameters assumed in the CSB safety analysis are protected. In addition, the CVDF process for mechanically sealing the MCO before transport to the CSB must ensure that the CSB assumptions are valid.

The K Basins safety analysis report must similarly protect the key interfaces identified relative to K Basin processes in Chapter A3.0. HNF-SD-WM-SAR-062, *K Basins Safety Analysis Report*, establishes MCO basket loading controls and overall MCO loading controls to ensure that the initial conditions and assumptions used in the CSB safety analyses are valid.

A5.8 REFERENCES


Annex A — Canister Storage Building


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March 2000
CHAPTER A6.0

PREVENTION OF INADVERTENT CRITICALITY
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Annex A — Canister Storage Building

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<td>Canister Storage Building</td>
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<tr>
<td>CVDF</td>
<td>Cold Vacuum Drying Facility</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>$k_{\text{eff}}$</td>
<td>effective neutron multiplication factor</td>
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<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
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<tr>
<td>MHM</td>
<td>multi-canister overpack handling machine</td>
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<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
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<td>SSC</td>
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A6.0 PREVENTION OF INADVERTENT CRITICALITY

A6.1 INTRODUCTION

This chapter evaluates the potential for criticality of spent nuclear fuel (SNF) during normal, abnormal, and accident conditions in the Canister Storage Building (CSB). It further describes the engineered features and administrative controls that give added assurance that a criticality event during handling and storage of SNF at the CSB is incredible. Chapter 6.0 of the SNF Project Final Safety Analysis Report (FSAR) provides an overall description of the SNF Project criticality prevention program.

The SNF received at the CSB is contained in baskets tiered vertically inside a multi-canister overpack (MCO). The MCO is moved to the CSB in a transportation cask from the Cold Vacuum Drying Facility (CVDF) in a nominally dry condition, except for a small amount of water or hydrogen chemically bound in the fuel matrix. MCOs will be stored at the CSB in an array of vertical tubes, two MCOs per tube, in a below ground vault that is designed for interim dry storage. The CSB facility design is described in detail in Chapters A2.0 and A4.0. CSB facility design aspects pertinent to criticality prevention are summarized in Section A6.4, which also specifies administrative controls.

Many SNF geometries, events, and conditions have been hypothesized for all operating and storage functions in the CSB under normal and accident conditions. The quantification of the approach to criticality for all of these actual and hypothesized conditions is described in HNF-SD-SNF-CSER-005, *Criticality Safety Evaluation Report for the Multi-Canister Overpack*. Many of the scenarios analyzed in HNF-SD-SNF-CSER-005 involving the MCOs and the fuel baskets are the same or similar for various facilities. A selected set of analyses of the bounding and credible conditions applicable to the CSB are described. These analyses establish the criticality safety design limits, their bases, and the parameters to be applied for prevention of a criticality in the CSB.

The scope of the analysis presented in this chapter includes only the SNF received at the CSB that has been loaded into MCOs at the K Basins and cold vacuum dried at the CVDF. Other fuels stored in the Interim Storage Area at the CSB are addressed in Annex D, “200 Area Interim Storage Area Final Safety Analysis Report.” The scope does not include handling of the SNF assemblies and the MCOs at the K Basins and the CVDF, or the transportation of the MCOs from the K Basins. Criticality concerns associated with loading and handling MCOs at the K Basins are addressed in HNF-SD-WM-SAR-062, *K Basins Safety Analysis Report*, while criticality issues associated with transporting the MCOs to the CSB are covered in HNF-SD-TP-SARP-017, *Safety Analysis Report for Packaging (Onsite) Multicanister Overpack Cask*. Unloading of the MCOs from the transportation cask and movement of the MCOs within the CSB are within the scope of this chapter. This chapter addresses the administrative controls at the K Basins to the extent that the failure of those administrative controls could affect criticality safety at the CSB.
The K Basins administrative controls and their failure modes are described in detail in HNF-2032, *SNF Fuel Retrieval Subproject Safety Assessment Document*.

This chapter also addresses implementation of requirements to achieve U.S. Nuclear Regulatory Commission equivalency in the design of the CSB with respect to criticality prevention based on the double contingency principle. The double contingency principle is stated in DOE Order 5480.24, *Nuclear Criticality Safety*, as follows: “Process designs shall incorporate sufficient factors of safety to require at least two unlikely, independent and concurrent changes in process conditions before a criticality accident is possible.” As described in HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements*, a system is considered acceptably subcritical if it has an effective neutron multiplication factor ($k_{eff}$) less than or equal to 0.95. The criteria used to determine whether $k_{eff}$ meets the 0.95 limit are described in Section 6.3.3.1 of the SNF Project FSAR. As applied in this chapter, criticality is a beyond double contingency event and is incredible.

The normal condition for the SNF at the CSB is in dry, sealed MCOs. The criticality calculations for normal conditions result in very low $k_{eff}$ values — less than 0.4. All credible conditions have calculated $k_{eff}$ values less than the limit of 0.95. It should be noted that although only a very small fraction of the fuel is unburned (<0.14%), all criticality analyses have been performed using fresh fuel compositions, which is conservative.

A bounding scenario was hypothesized to determine the maximum $k_{eff}$ values computed for Mark IA and Mark IV fuel in an MCO. This scenario involves a drop accident that results in the following:

- The shipping cask breaches
- The MCO breaches
- Hydrogen is released
- An ignition source is present
- Hydrogen ignites and the Hanford Fire Department responds
- The geometry is distorted
- All fuel rubblizes to optimally sized and optimally spaced pieces
- Moderator is added to the interior of the MCO during fire fighting.

Furthermore, the MCO was assumed to contain a mass distribution of unirradiated fuel that maximizes the calculated $k_{eff}$. Even under these hypothetical conditions, the system was still not critical. The maximum $k_{eff}$ computed was approximately 0.97 for a Mark IA MCO and approximately 0.93 for a Mark IV MCO.

For $k_{eff}$ of an MCO or an array of MCOs to exceed 0.95 independent of controls requires not only the failure of design features and the failure of administrative controls but also a sequence of hypothetical conditions that must occur concurrently. These conditions are incredible, as discussed below.
Fuel rubblization into optimally sized particles is judged to be incredible because when discharged from the N Reactor, the individual fuel elements survived vertical drops of more than 40 feet with only minor damage, and the elements also will have been processed by the primary clean machine in the K Basins without rubblizing.

Optimization of the fuel requires that it form pieces that are approximately twice the diameter of the thickness of the original fuel elements, which is judged to be incredible.

Rubblized fuel and scrap achieving and maintaining optimum spacing is judged to be incredible because there is nothing to support the fuel particles in an optimum matrix.

In conclusion, the $k_{eff}$ for all credible scenarios is considerably less than 0.95 (HNF-SD-SNF-CSER-005).

**A6.2 REQUIREMENTS**

The requirements that form the basis of criticality prevention are identified in Section 6.2 of the SNF Project FSAR. Technical safety requirements for nuclear criticality safety at the CSB specify that a program will be established, implemented, and maintained to prevent an accidental criticality at the CSB. Criticality limits and controls will be documented in criticality safety evaluation reports and implemented in criticality prevention specifications.

**A6.3 CRITICALITY CONCERNS**

**A6.3.1 Criticality Hazards**

See Section 6.3.1 for identification of the fissile material available in SNF Project facilities and loaded into MCOs. The CSB is classified as a limited control facility because it contains greater than one-third of a minimum critical mass.

**A6.3.2 Criticality Analysis**

The criticality analyses for the CSB consider only fissionable material contained in MCOs.

**A6.3.2.1 Multi-Canister Overpack Loadings.** Baskets containing fuel or scrap are loaded into MCOs at the K Basins. An MCO can contain either six Mark IA fuel and scrap baskets or five Mark IV fuel and scrap baskets. Normal MCO loadings and limits are described in HNF-SD-WM-SAR-062. Each MCO may be loaded in one of three normal configurations:

- All fuel baskets
- One scrap basket and four or five fuel baskets
Two scrap baskets and three or four fuel baskets.

Mark IA and Mark IV baskets are not loaded into the same MCO. It is permissible to mix Mark IV fuel or scrap with Mark IA fuel or scrap in Mark IA baskets. As described in HNF-SD-WM-SAR-062, the only situation in which Mark IA fuel would be loaded into Mark IV baskets occur when the long-length (26-in.) Mark IA fuel assemblies in the K West Basin are loaded into Mark IV fuel baskets.

A6.3.2.2 Description of the Canister Storage Building. The CSB is composed of an operating area and a below grade SNF storage area consisting of three vaults. Storage tubes for MCOs are located in vault 1 only. There are 220 tubes arranged in a 10 (north-south) by 22 (east-west) hexagonal array. The tube center-to-center distance is 4 ft, 8 in. in the east-west direction and 4 ft, 6 in. in the north-south direction. Two normal MCOs are placed in each storage tube in a vertical column, as shown in Figure A6-1. There is an impact absorber at the bottom of each tube and between the two MCOs in a tube. The bottom impact absorber is 36 in. long, and the intermediate impact absorber is 24 in. long (see Figure A2-13). Further details are provided in Chapters A2.0 and A4.0.

Six additional overpack storage tubes are also available in vault 1 for storing damaged MCOs or MCOs suspected of leakage. The overpack storage tubes are located along the west edge of the 10-by-22 array of normal MCO storage tubes as shown in Figure A2-7. There are no provisions for storing more than one MCO in an overpack storage tube. Thus, there is only one impact absorber at the bottom of each overpack storage tube.

The vault walls, vault basemat, and operating deck are constructed of reinforced, high-density concrete (see Chapter A2.0 for details). The exterior vault walls are 4.5 ft thick and the walls between vaults are 3 ft thick. The vault basemat is 5.5 ft thick and the operating deck is 5 ft thick. The thick concrete walls between the vaults neutronically isolate the fissile material in one vault from that in adjacent vaults. Therefore, reactivity of the CSB may be assessed by analyzing a single vault.

As described in Chapter A4.0, the CSB substructure consists of safety-class components designed to prevent failure of MCOs during the natural phenomena hazards listed in Chapter A3.0, and to maintain the SNF in a subcritical geometry. Therefore, complete collapse of the vault storage tubes, resulting in rearrangement of the MCOs in the vault, is considered a beyond design basis contingency and was not analyzed.

While at the CSB, an MCO is in its transportation cask; in the MCO handling machine (MHM); in a sampling/weld station; or in a storage tube. The MHM, which has a bell-shaped configuration with 10-in.-thick stainless steel walls, is used to remove the MCO from its cask and to move the MCO to and from process pits in the sampling/weld area and storage tubes in the vault. There is a row of seven process pits in the sampling/weld area, but only two of the pits are needed. However, there are no criticality concerns with using more than two pits. Two of the
remaining pits provide utilities for the sampling/weld stations. The other three pits are to be used for future projects.

A6.3.2.3 Criticality Hazard Configurations. Table A6-1 identifies the normal operating conditions considered, and for each, the hypothetical criticality hazards that could be created by abnormal occurrences during MCO handling and storage at the CSB. Also listed are the features that prevent or mitigate the consequences of each abnormal occurrence. The identified criticality hazards are based primarily on the hazard analysis, HNF-SD-SNF-HIE-001, Canister Storage Building Hazard Analysis, which is summarized in Chapter A3.0. Table A6-1 provides information for selecting credible criticality scenarios and for determining the scope of the required criticality ($k_{\text{eff}}$) analyses.

A6.3.3 Criticality Models and Calculations

A complete description of the models used and calculations completed for the criticality analyses is given in HNF-SD-SNF-CSER-005. A description of the codes used to perform the criticality analyses, their modeling assumptions, and their validation is also provided in HNF-SD-SNF-CSER-005. A variety of models was used in the criticality analyses, due in large part to the fact that the analyses were in progress while MCO and CSB designs were still evolving. Not all calculations were redone to incorporate the latest design parameters. Typically, a calculation was not redone if it was still bounding or if it was used only in a sensitivity study and was still adequate for that purpose. Also, calculations were not rerun for highly subcritical conditions if the design changes were not major. In such cases, representative results were deemed sufficient.

The effect of uncertainties such as moisture content, fuel length selections, and corrosion were investigated by performing calculations over the range of possible variations for the individual parameters (HNF-SD-SNF-CSER-005). These calculations showed that $k_{\text{eff}}$ typically increases with water addition. Both corrosion and fuel relocation (e.g., settling) reduce $k_{\text{eff}}$ by reducing the amount of either fuel or water or by producing a less optimal geometry. The longest elements for the Mark IV fuel assemblies and the 20.9-in.-long elements for the Mark IA fuel assemblies were chosen to represent “normal conditions.” A study of fuel length versus the $k_{\text{eff}}$ of a flooded MCO concluded that there is a modest sensitivity in $k_{\text{eff}}$ due to changes in fuel length (HNF-SD-SNF-CSER-005, Appendix B). However, use of this length to represent normal conditions is justified because about 88% of the Mark IA fuel is 20.9 in. long and use of the optimal length (19.6 in.) increases $k_{\text{eff}}$ by only 0.006.

MCOs at the operating deck level of the CSB are handled as single units located inside the MHM. There is no credible mechanism for placing two MCOs side-by-side outside their transportation casks on the CSB operating deck. As described in Chapter A2.0, an MCO is in its transportation cask, in the MHM, in a sampling/weld station, or in a storage tube. Therefore, all criticality scenarios on the operating deck were analyzed for a solitary MCO.
<table>
<thead>
<tr>
<th>Configuration number</th>
<th>Location or activity</th>
<th>Normal operating conditions</th>
<th>Abnormal or accident conditions (contingencies)</th>
<th>Preventive features</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Cask–MCO receiving area</td>
<td>Dry MCO in a dry cask (see note 1)</td>
<td>Water inside the MCO or the transportation cask; flooding all around (see note 2)</td>
<td>Receipt and acceptance of a flooded cask–MCO at the CSB is prevented by administrative controls at the CVDF and CSB. There are no sources of water or other moderators around the CSB operating area or storage vault except as noted at the sampling/weld area.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Overloaded MCO (see note 2)</td>
<td>Fuel basket and scrap basket overloading are prevented by administrative controls at the K Basins.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Misloaded MCO (see note 2)</td>
<td>Administrative controls at the K Basins</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Collapse of CSB structures due to an earthquake or other causes (see note 2)</td>
<td>Safety-class SSCs at the CSB; safety-class MCO shell and shield plug</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Cask–MCO drop from receiving crane; if MCO is breached, some of the contents could spill out or a hydrogen burn might occur.</td>
<td>Safety-class MCO shell and shield plug protect initial assumptions for geometry control. The CSB pre-fire plan restricts the use of water to fight fires (see Chapter A11.0).</td>
</tr>
<tr>
<td>2</td>
<td>Lifting MCO from transportation cask into MIM</td>
<td>Dry MCO in dry MIM</td>
<td>MCO shear or drop; if MCO is breached, some of the contents could spill out or a hydrogen burn might occur.</td>
<td>Safety-class MCO shell and shield plug protect initial assumptions for geometry control. The CSB pre-fire plan restricts the use of water to fight fires (see Chapter A11.0).</td>
</tr>
<tr>
<td>3</td>
<td>Placing MCOs in sampling/weld station</td>
<td>Up to two dry stations contain one dry MCO each</td>
<td>MCO dropped into a station</td>
<td>Safety-class MCO shell and shield plug protect initial assumptions for geometry control.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>MCO placed on the top of another MCO in a pit</td>
<td>None required</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>MCOs in one or more unused pits</td>
<td>None required</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Possible loss of MCO containment function</td>
<td>Safety-class MCO shell and shield plug protect initial assumptions for geometry control.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Glycol–water from shield cooling system leaks resulting in coolant inside and around an MCO</td>
<td>Safety-class MCO shell and shield plug protect initial assumptions for geometry control. Quantity of coolant is limited.</td>
</tr>
</tbody>
</table>
## Table A6-1. Criticality Hazard Configurations. (2 sheets)

<table>
<thead>
<tr>
<th>Configuration number</th>
<th>Location or activity</th>
<th>Normal operating conditions</th>
<th>Abnormal or accident conditions (contingencies)</th>
<th>Preventive features</th>
</tr>
</thead>
<tbody>
<tr>
<td>4</td>
<td>Moving MCO in MHIM and placing in a predetermined storage tube</td>
<td>Dry MCO in dry storage tube with two MCOs per tube; intermediate impact absorber in place between MCOs</td>
<td>MCO dropped into the tube, colliding with the bottom impact absorber or intermediate impact absorber</td>
<td>Safety-class MCO shell and shield plug protect initial assumptions for geometry control.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>MCO shear while being lowered into storage tube</td>
<td>Safety-class MCO shell and shield plug protect initial assumptions for geometry control.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Possible loss of MCO containment function</td>
<td>Safety-class MCO shell and shield plug protect initial assumptions for geometry control.</td>
</tr>
<tr>
<td>5</td>
<td>Interim MCO storage in tubes</td>
<td>A 10 (north-south) by 22 (east-west) array with two dry MCOs per standard dry storage tube; storage tubes normally vented; up to six additional MCOs stored in the overpack storage tubes (one MCO per tube)</td>
<td>Water intrusion inside storage vault, storage tubes and MCO.</td>
<td>CSB tube structure maintains the storage array.</td>
</tr>
</tbody>
</table>

**Note 1:** "Dry" means only a minimal quantity of water.

**Note 2:** Abnormal or accident events involving CSB and MCO flooding, MCO misloading and overloading, and earthquakes are common to all five configurations. In general, they are listed only once, for configuration 1, but apply to the other configurations as well.

CSB = Canister Storage Building.
CVDF = Cold Vacuum Drying Facility.
MCO = multi-canister overpack.
MHIM = multi-canister overpack handling machine.
SSC = structure, system, and component.
Six calculational models were employed in the criticality analyses. The first three models simulate various locations for the MCO outside the storage vault. The other three models simulate the CSB vault area where the SNF will be stored. The MCOs in all models can be either water-flooded or dry. MCOs are normally dry at the CSB; however, flooded MCOs were analyzed as a contingency.

1. Figure A6-2 shows the first model, which represents an MCO inside the MHM on the operating deck. The operating deck's concrete floor acts as a neutron reflector. This model bounds the transportation cask-MCO configuration, which has a maximum of 8 in. of steel shielding.

2. Figure A6-3 shows the second model, which depicts an MCO surrounded by concrete. This model was used to evaluate a normal configuration in which an MCO is lowered through the concrete operating deck into a storage tube.

3. The third model represents an MCO in a sampling/weld station where there is steel shielding around the MCO and an impact absorber and concrete below it. To be conservative, the impact absorber was omitted from this model. As shown in Figure A6-4, there is an annulus filled with coolant (a propylene glycol and water mixture) in the steel surrounding the MCO. The annulus is 1.75 in. thick and is 2.75 in. outside the MCO. The steel between the MCO and the coolant represents the wall of the sampling/weld station pit, and the steel outside the coolant represents the rotating shield (see Chapter A2.0 for details of the sampling/weld station design). The weld cap that also contains coolant was not included in the model. However, the MCO shield plug isolates the coolant from the fissionable material inside the MCO.

4. The fourth model, shown in Figure A6-5, represents a 10-by-22 array of normal storage tubes with two MCOs per tube. The impact absorbers in the tubes are modeled as cylinders containing steel at 50% density. The bottom impact absorbers are 36 in. long, while the intermediate impact absorbers are 24 in. long. The MCO basket loadings in this model, which are based on the latest design specifications, are described in Section 6.3.2 of the SNF Project FSAR. The six overpack storage tubes for MCOs are not included in this model, however this modeling simplification has a negligible effect on reactivity. There is only one MCO per overpack tube, and the overpack tubes are spaced further apart than the normal tubes, as shown in Figure A2-7.

5. The fifth model is an earlier version of the fourth model described above. The key differences between the two models are that the fifth model has air in place of the impact absorbers, has concrete boundaries more closely surrounding the storage tube array and immediately above the upper MCO, and has scrap baskets loaded higher than the current limit allows.

6. Figure A6-6 shows the sixth model, which depicts the storage tube matrix modeled as an infinite array. The axial detail in this model is the same as in the fifth model.
The fifth model, like the fourth, neglects the overpack tubes, while the sixth model contains an infinite number of normal tubes. Thus the fifth and sixth models bound the case in which there are MCOs in the six overpack storage tubes. Although the fourth model described above (see Figure A6-5) is the latest and most closely simulates the current design of the MCO, its loadings, and the vault storage tubes, not all calculations of the vault area were repeated using this model. The sixth model, with the infinite array of tubes (see Figure A6-6), proved to give the most conservative results (highest $k_{eff}$ values) where comparative values were available. These results were still below the limit of 0.95 (HNF-SD-SNF-CSER-005). Thus, some of the results reported in the next section were based on the infinite array model and are bounding.

The configurations analyzed using the models described above bound all normal and hypothetical abnormal or accident conditions. Details of the modeling of fuel elements and fuel scrap inside the MCOs are described in HNF-SD-SNF-CSER-005.

Using the six analysis models, criticality calculations were performed for normal and abnormal or accident conditions for MCOs containing N Reactor fuel (Mark IV and Mark IA) and fuel scrap. Under normal conditions, the N Reactor fuel was assumed to be loaded in the MCOs as described in Section A6.3.2.1. That is, the top and bottom baskets were assumed to contain scrap, and the other baskets were assumed to contain intact fuel assemblies, which give the highest $k_{eff}$ for a particular fuel type. Fuel baskets can also contain damaged fuel assemblies (i.e., partial assemblies, individual elements, or pieces of elements). However, any possible configuration of damaged fuel in an MCO is bounded by cases of an MCO containing optimally rubblized and moderated fuel and optimal scrap. Such optimized MCO loadings have been analyzed and found to be critically safe (i.e., $k_{eff} < 0.95$) under the hypothetical flooding scenario analyzed (see HNF-SD-SNF-CSER-005 and Section A6.4.4). The specific cases analyzed, based on the conditions and features identified in Table A6-1, are described in Sections A6.3.4.1, A6.3.4.2, and A6.3.4.3, along with the results obtained.

Under normal conditions, the Mark IA and Mark IV fuel baskets are to be fully loaded (48 Mark IA assemblies in the Mark IA baskets, or 54 Mark IV assemblies in the Mark IV baskets). Fully loaded baskets are slightly undermoderated even when flooded with water to the maximum extent possible. An extensive study was done to determine the most reactive fuel basket loadings when flooded with water (HNF-SD-SNF-CSER-005). That study concluded that the most reactive MCO loadings consist of two scrap baskets plus three Mark IV or four Mark IA partially loaded fuel baskets. For the most reactive Mark IA MCO, the four fuel baskets each contain 47 fuel assemblies with one empty location in the middle row (see Figure 6-4 in the SNF Project FSAR). For the most reactive Mark IV MCO, the three fuel baskets each contain 53 complete fuel assemblies with only an inner element in one outer row location (see Figure 6-5 of the SNF Project FSAR). Unless otherwise noted, the results given in Section A6.3.4 are based on these MCO partial loadings. The $k_{eff}$ values of the most reactive loadings are only about 0.002 higher than those of full basket loadings.

As fuel assemblies were removed from the fully loaded models used in the study (HNF-SD-SNF-CSER-005), the remaining assemblies were not redistributed to optimize the lattice spacing because the assemblies are held in fixed locations within the fuel baskets by design.
Thus, only those assemblies adjacent to a removed assembly benefited from the increased moderation. Under credible accident scenarios, the fuel could move within the baskets. However, the reactivity effects of such movement are bounded by cases in which the fuel is reduced to rubble.

The scrap baskets modeled at each end of the MCO contained optimized fuel scrap. The uranium in the Mark IV scrap baskets was 0.95 wt% $^{235}$U (the same as in Mark IV fuel), while the uranium in the Mark IA scrap baskets was 1.25 wt% $^{235}$U. The size and spacing of the scrap were adjusted to give the highest $k_{\text{eff}}$ in a water-moderated environment — the most limiting condition (HNF-SD-SNF-CSER-005). The optimized scrap is thus more reactive under water moderation than intact fuel or accumulated corrosion products (canister sludge).

Loading intact long-length Mark IA fuel assemblies described in Section A6.3.2.1 from the K West Basin into a Mark IV fuel basket was determined to result in only a small, statistically insignificant increase in $k_{\text{eff}}$ compared to a typical Mark IV fuel basket loading (HNF-SD-SNF-CSER-005). That analysis showed this to be true regardless of the row and basket into which the long-length assemblies are placed. Because there will be only one basket with long-length Mark IA assemblies and these assemblies do not increase $k_{\text{eff}}$ significantly, the results reported in Section A6.3.4 are based on the typical Mark IV fuel basket loadings described above.

A small quantity (about 0.25 metric tons) of Mark IA fuel and scrap is stored in the K East Basin. This material will be loaded into Mark IA fuel and scrap baskets with Mark IV fuel and scrap subject to limits specified in HNF-SD-SNF-CSER-010, *Criticality Safety Evaluation Report for Storage and Removal of Spent Nuclear Fuel in the K Basins*.

### A6.3.4 Criticality Analysis Results

This section presents the results of the analyses performed to evaluate the potential for criticality under normal and abnormal or accident conditions, and provides bases for establishing the criticality safety design limits, the requirements for engineered design features discussed in Sections A6.4.1 and A6.4.2, and the administrative controls for criticality prevention discussed in Section A6.4.3. The results also verify compliance with the double contingency requirement during storage and handling of the SNF contained in the MCOs at the CSB because a criticality has been shown to be incredible. As stated earlier, the principal criterion for prevention of criticality as applied in the analyses is that $k_{\text{eff}}$ must not exceed 0.95 for any single credible contingency. The codes used to perform the criticality analyses, their modeling assumptions, and their validation are described in HNF-SD-SNF-CSER-005.

The criticality analyses are divided into two main parts: normal (dry) conditions and abnormal or accident conditions. The results demonstrate that in the dry condition, the value of $k_{\text{eff}}$ is well below 0.95 even if the fuel geometry is distorted in a postulated accident or if the fuel is misloaded into the wrong baskets (e.g., Mark IA fuel or scrap in Mark IV baskets). Scenarios
involving water inside the MCOs or flooding of vaults concurrent with other abnormal or accident conditions are considered beyond design basis accidents for the CSB.

A6.3.4.1 Normal Conditions. All MCOs received at the CSB have been drained and vacuum dried at the CVDF. The interiors of the MCOs are thus dry except for approximately 3 kg of water, including free water, hydrides, hydrates, and hydroxides, chemically bound in the fuel matrix. Under normal conditions, water or other moderating materials are not added to the MCOs while the MCO is at the CSB. LA-12808, Nuclear Criticality Safety Guide, states that low-enriched uranium (<5 wt% $^{235}$U) cannot be made critical without internal moderation. Analyses documented in HNF-SD-SNF-CSER-005 demonstrate that unmoderated MCOs have $k_{eff}$ values less than 0.4. Since MCOs from the K Basins are received and maintained in a dry state at the CSB, and since they will not contain uranium enriched to more than 1.25 wt% $^{235}$U, a criticality accident while handling and storing these MCOs at the CSB is impossible under normal conditions. Fuels with enrichments over 1.25 wt% $^{235}$U and fuels with mixed plutonium and uranium oxide are stored in the Interim Storage Area at the CSB. Criticality issues associated with these fuels are addressed in Annex D.

A6.3.4.2 Abnormal or Accident Conditions — Loss of Engineered and Administrative Controls during Handling and Storage. The two primary criticality controls at the CSB are the geometry and the absence of moderator. As stated above, drained MCOs containing low-enriched uranium (<5 wt% $^{235}$U) cannot be made critical without internal moderation. Therefore, the CSB operating area is considered a moderator control area in accordance with ANSI/ANS-8.22-1997, American National Standard for Nuclear Criticality Safety Based on Limiting and Controlling Moderators. The only control on moderator that is implemented is for fire fighting. There is no automatic fire suppression system or other source of water in the operating area of the CSB.

The CSB is designed to withstand a design basis earthquake and prevent flooding without loss of its safety function. The only significant neutron moderator normally found in the CSB is in the sampling/weld station, which has a cooling system for the shielding around the upper portion of the MCO and the cover cap. A failure of the cooling system that allows the coolant (a mixture of propylene glycol and water) to leak into an MCO through ports in the shield plug, which are normally closed, is an unlikely event. The volume of coolant in the cooling system is approximately 80 gal; approximately 130 gal of space is available in a full MCO. Thus coolant could flood only the lower two-thirds of the MCO at most. Analysis of the draining of an MCO at the CVDF (HNF-SD-SNF-CSER-005 and HNF-3553, Annex B) showed that a fully flooded MCO bounds all partially filled configurations.

It is also possible, but unlikely, that a flooded MCO could be received and accepted at the CSB. This contingency is prevented by administrative controls at both the CVDF and the CSB. Associated documentation will be examined to verify that the MCO was drained and dried at the CVDF. Other moderators would have to be purposely brought into the facility.

Fire fighting is the only credible reason for deliberately bringing a significant quantity of moderating material into the CSB. Water, by itself, is not a problem at the CSB because the
MCOs are fully flooded when loaded and transported from the K Basins. In addition, the facility pre-fire plan, described in Chapter A11.0, restricts the use of water to fight a fire in the CSB operating area. The pre-fire plan and the requirement to develop a recovery plan, should the drop of an MCO occur, satisfy the requirements of ANSI/ANS 8.22-1997 and HNF-PRO-547, *Criticality Safety for Firefighting*, which specifies the methods to be used in fighting a fire involving fissionable material.

As discussed in Section A6.1, the sequence of events that could lead to loss of geometry control and cause $k_{\text{eff}}$ to exceed 0.95 is incredible. Furthermore, as the Mark IA fuel and scrap settle to a higher-than-optimal packing fraction following the postulated MCO drop and the hypothetical fuel rubblization, maximum potential $k_{\text{eff}}$ rapidly drops below 0.95, as shown in Figure A6-7. This settling will occur shortly after the drop accident, well before any fire fighting activity could introduce moderator. All follow-up actions would be closely controlled by the recovery plan, which would specify the means of fire suppression should a fire occur. Solid streams of water could only be used as a last resort.

It is thus unlikely that the $k_{\text{eff}}$ limit would be exceeded even if the type of drop and hypothetical consequences described above occurred. Therefore, the CSB is designated a Category B fire fighting area.

The conclusion of the evaluation in SNF-4042, *Evaluation of Accident Frequencies at the Canister Storage Building*, is that the expected frequency of a drop accident that results in a breached MCO and cask, an ensuing fire, and the accidental introduction of a significant quantity of water to the exposed SNF is less than $10^{-5}$/yr, making the postulated events incredible.

Credible (single contingency) scenarios for having moderated SNF at the CSB are then (1) receiving and accepting a water-flooded MCO and (2) partial flooding of an MCO with a mixture of propylene glycol and water at the sampling/weld station because of a leak in the system designed to cool the shielding around the upper portion of the MCO and the cover cap. The second scenario is bounded by the first, so only the first was analyzed as described below.

Cases gly4.3, mhm4.9, and oc2.12c in Table A6-2 represent a scenario in which a flooded MCO is received at the CSB, inadvertently bypassing the CVDF. This event could lead to four geometries that were analyzed for criticality potential. The $k_{\text{eff}}$ values computed for all configurations were below the limit of 0.95.

1. A flooded Mark IV MCO is located in the sampling/weld station. It was assumed that there was also a leak in the cooling system and a mixture of propylene glycol and water flooded the annular cooling channel in the radial shielding surrounding the MCO in the pit. The resulting $k_{\text{eff}}$ was 0.926 for case gly4.3 in Table A6-2. A similar calculation for a Mark IA MCO yielded a lower $k_{\text{eff}}$ of 0.901 (case gly1.3 in HNF-SD-SNF-CSER-005).
### Table A6-2. Canister Storage Building Criticality Analysis Results for Flooding Scenarios.

<table>
<thead>
<tr>
<th>Case</th>
<th>Description*</th>
<th>Modelb</th>
<th>Water density (g/cm³)</th>
<th>$k_{calc}$</th>
<th>Standard deviation ($\sigma_k$)</th>
<th>$k_{eff}$ (95% CL with biases)c</th>
<th>Remarks</th>
</tr>
</thead>
<tbody>
<tr>
<td>gly4.3</td>
<td>Mark IV fuel in a flooded MCO at the sampling/weld station, which is flooded with propylene glycol and water</td>
<td>3</td>
<td>MCO - 1.0</td>
<td>0.915</td>
<td>0.0009</td>
<td>0.926</td>
<td>One MCO with steel around it and concrete below; the steel shielding includes an annular region containing a propylene glycol and water coolant</td>
</tr>
<tr>
<td>oc2.12c</td>
<td>Mark IV fuel, two MCOs per tube; MCOs flooded</td>
<td>6</td>
<td>MCO - 1.0 Tube - 0.008 Vault - 0.008</td>
<td>0.903</td>
<td>0.0022</td>
<td>0.914</td>
<td>Infinite tube lattice is bounding</td>
</tr>
<tr>
<td>mh4m.7</td>
<td>Mark IV fuel, one flooded MCO in MHM</td>
<td>1</td>
<td>MCO - 1.0 No water outside MCO</td>
<td>0.918</td>
<td>0.0012</td>
<td>0.930</td>
<td>10-in. steel walls around and above the MCO and 5-ft-thick concrete below it</td>
</tr>
<tr>
<td>mh4m.9</td>
<td>Mark IV fuel, one flooded MCO surrounded by concrete</td>
<td>2</td>
<td>MCO - 1.0 No water outside MCO</td>
<td>0.919</td>
<td>0.0012</td>
<td>0.930</td>
<td>One MCO with concrete around it and steel above it</td>
</tr>
</tbody>
</table>

* MCO loadings for the cases listed in this table contain Mark IV fuel and scrap only. One Mark IV fuel basket will contain 12 long-length Mark IA assemblies as described in Sections A6.3.2 and A6.3.3. These assemblies increase $k_{eff}$ over the values listed, but the differences are not statistically significant.

A small quantity of Mark IA fuel and scrap stored in the K East Basin also will be loaded into Mark IV baskets. Allowed loadings of this material (HNF-SD-SNF-CSER-010) are bounded by analyzed fuel misloading cases, which had $k_{eff}$ values less than 0.95 (HNF-SD-SNF-CSER-005).

* $k_{eff} = k_{calc} + 0.0004 + 0.01 + 1.645^2 (\sigma_k^2 + 0.002083^2)$ as defined in Section 6.3.3 of the SNF Project FSAR. This value must not exceed the criticality prevention criterion of 0.95.

* This MCO loading was not based on the latest design data but is very similar. Fuel baskets contain a full load, which is slightly less reactive (by $\sim 0.002 \Delta k_{eff}$) than optimally loaded fuel baskets. The two scrap baskets, which dominate the reactivity of a flooded MCO, have slightly different dimensions and loadings than those used in the latest models. They contain 7% more fuel mass than the current limit of 2,156 lb but have a $k_{eff}$ comparable to optimized loadings that adhere to the limit.

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CL = confidence level.
MCO = multi-canister overpack.
MHM = multi-canister overpack handling machine.
2. A flooded Mark IV MCO is housed in the MHM, which is modeled as having 10 in. of steel all around the MCO except at the bottom where 5 ft of concrete represents the operating deck. The space between the outer wall of the MCO and the inner wall of the MHM and the environment outside the MHM are assumed to be dry. The resulting $k_{\text{eff}}$ is 0.930, as shown for case mhm4.7 in Table A6-2. A similar calculation for a Mark IA MCO yielded a lower $k_{\text{eff}}$ of 0.910 (case mhml.7 in HNF-SD-SNF-CSER-005).

3. A flooded Mark IV MCO is being lowered through the concrete floor into a storage tube. The configuration is simulated by using a model of a flooded MCO surrounded radially by concrete with 10 in. of steel above it. The resulting $k_{\text{eff}}$ is 0.930, as shown for case mhm4.9 in Table A6-2. A similar calculation for a Mark IA MCO yielded a lower $k_{\text{eff}}$ of 0.908 (case mhml.9 in HNF-SD-SNF-CSER-005).

4. A flooded Mark IV MCO is located in a storage tube. This configuration is simulated by the bounding case in which all 440 MCOs in the storage vault are flooded. The resulting $k_{\text{eff}}$ is 0.914, as shown for case occ2.12c in Table A6-2. A similar calculation for flooded Mark IA MCOs in the vault yielded a lower $k_{\text{eff}}$ of 0.905 (case occ1.15c in HNF-SD-SNF-CSER-005).

As noted in Table A6-2, the water inside the MCOs was modeled with a density of 1.0 g/cm$^3$ (full flooding) while water assumed to be present in the empty vault spaces outside the MCOs was modeled with a density of 0.008 g/cm$^3$. Both densities were based on sensitivity studies that indicated they resulted in the highest values of $k_{\text{eff}}$ (HNF-SD-SNF-CSER-005).

In the sensitivity analysis, water density in and between the storage tubes was varied from 0.0 g/cm$^3$ to 1.0 g/cm$^3$ using the model shown in Figure A6-6. Water density in empty vault spaces outside the MCOs had a uniform spatial distribution in each case. Conditions inside the MCOs, which were modeled as containing Mark IV fuel, were assumed to be dry (water density of 0.005 g/cm$^3$). Also, the model used did not include the latest MCO loading design.

Each of the three fuel baskets in the MCO contained 54 intact fuel assemblies. The two scrap baskets in each MCO had slightly different dimensions and loadings than those used in the latest models. Consequently they contained about 7% more fuel mass than the latest models but had a $k_{\text{eff}}$ comparable with optimized loadings that adhere to the fuel mass limit of 2,156 lb specified in HNF-SD-SNF-CSER-010.

The results of the study, shown graphically in Figure A6-8, indicate that a water density of 0.008 g/cm$^3$ outside the MCOs yields the peak value for $k_{\text{eff}}$. Increasing the water density between MCOs above 0.008 g/cm$^3$ decouples the MCOs by increasing neutron captures in water, resulting in a reduction of the $k_{\text{eff}}$. With the MCOs flooded, $k_{\text{eff}}$ is much less sensitive to variations in water density outside the MCOs (HNF-SD-SNF-CSER-005).

A6.3.4.3 Abnormal or Accident Conditions — Misloaded Multi-Canister Overpacks.

Scenarios stemming from loss of administrative controls at the K Basins also were postulated.
These scenarios, which all involve fuel basket misloading at the K Basins, are described below. Only dry MCOs are considered because a flooded MCO at CSB requires both the receipt and acceptance of a flooded MCO at the CSB. Thus, no credible MCO misloading at the K Basins can result in a criticality accident at the CSB without the additional contingency of water flooding. Analysis of multiple-contingency accidents is discussed in Section A6.4.4.

1. **One additional scrap basket misloaded into MCO.** As stated in Section A6.3.2.1, the criticality safety analyses assume that two scrap baskets are loaded in each MCO, one scrap basket being on the top of the stack and the other at the bottom. The fuel baskets in the middle of the MCO are loaded with intact fuel assemblies of the same enrichment as the scrap. In this misloading scenario, it was assumed that a third scrap basket was loaded at the second from the top position. This configuration places the misloaded scrap baskets in positions that maximize $k_{en}$ (HNF-SD-SNF-CSER-005).

2. **A canister of Mark IA scrap misloaded into a Mark IV scrap basket.** In this scenario, the Mark IV scrap basket at the top of the MCO was assumed to be misloaded with a canister (233 kg) of Mark IA scrap (1.25 wt% $^{235}\text{U}$). The remainder of the basket was correctly loaded with Mark IV scrap (0.95 wt% $^{235}\text{U}$). A similar scenario is that of a Mark IA intact fuel canister misloaded into a Mark IV fuel basket. However, because scrap baskets were modeled as containing optimized arrangements of fuel and water and are more reactive than intact fuel baskets (HNF-SD-SNF-CSER-005), this scenario is bounded by the scenario involving a misloaded scrap basket.

The two MCO misloading scenarios were shown to bound all other credible misloading scenarios at the K Basins (HNF-SD-SNF-CSER-005). The first scenario represents the worst-case misloading of a basket into an MCO, and the second scenario represents the worst-case misloading of a canister into a basket. However, both cases are highly subcritical without the addition of water.

### A6.4 CRITICALITY CONTROLS

#### A6.4.1 Passive Engineering Controls

The engineering controls for preventing inadvertent criticality in the CSB include facility structures, features that prevent introduction of water into the vault area, and MCO structures. As described in Chapters A2.0 and A4.0, the CSB structures, including MCO storage tubes, are designed to perform the safety function of providing support for MCOs to maintain their structural integrity before, during, and following all design basis accidents, including the design basis earthquake. The following passive features are designated as safety-class to maintain the MCOs in a critically favorable array:

- Standard and overpack storage tubes
- Tube base assemblies.

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Annex A — Canister Storage Building
There is no natural groundwater source in the immediate vicinity of the vault walls and floor. The water table in the 200 East Area is more than 200 ft below the bottom of the vault basemat. Area drainage systems route rainwater away from the CSB.

There are no sprinkler systems required or provided for fire protection in the CSB that could flood the operating floor, the sampling/weld stations, or the vault area (see Chapter A2.0). There is no water piping provided adjacent to or above the CSB vaults. However, as described in Chapter A2.0, rooms in the support building, located at the north end of the CSB, are protected by a wet-pipe sprinkler system. A buried 6-in. fire water line running from a fire water main loop to the northwest corner of the support building feeds this sprinkler system.

Safety-class features of the MCO and the Mark 1A baskets provide geometry control sufficient to ensure criticality safety should a flooded MCO be received at the CSB. The analysis that demonstrates this is discussed in Section A6.3.4.2, and the results are summarized in Table A6-2. The following geometry control features of the MCO and Mark 1A baskets have been engineered as safety class according to HNF-SD-SNF-SARR-005, Multi-Canister Overpack Topical Report:

- MCO shell
- Filter guard plate
- Shield plug
- Mark 1A fuel and scrap basket base plate
- Mark 1A fuel and scrap basket center post.

A6.4.2 Active Engineering Controls

There are no active engineering controls required at the CSB to ensure criticality safety.

A6.4.3 Administrative Controls

MCOs arriving at the CSB will have been subject to controls in the K Basins and CVDF to ensure that specifications are met for MCO loading, draining, and drying. Control of operations performed before the MCO is received at the CSB is discussed in Annex B, the CVDF FSAR, and in the K Basins FSAR (HNF-SD-WM-SAR-062). CSB personnel inspect MCO documentation to verify the acceptability of an MCO upon receipt. Verification before acceptance of each MCO delivery ensures the proper processing steps have been carried out and proper conditions met for each MCO. Recovery planning activity for out-of-specification MCOs that arrive at the CSB also is addressed. The disposition of such MCOs will depend on the potential consequences and related risks and on which corrective actions are most appropriate. The CSB criticality safety evaluation was based on the implementation of this program and related controls at the K Basins and the CVDF.
Criticality analysis results discussed in Section A6.3.4.3 show that a failure of administrative
controls at the K Basins, such as misloading a single canister of Mark IA fuel assemblies or scrap
in Mark IV fuel baskets, will not create a criticality concern in the CSB (i.e., $k_{eff}$ remains well
below the criticality safety limit for all credible accidents). Likewise, neither an abnormal event at
the CSB nor the loss of administrative controls at the CVDF that results in a flooded MCO being
received at the CSB will create a criticality concern, as discussed in Section A6.3.4.2. Procedures
will control the introduction of moderator into the CSB to comply with ANSI/ANS-8.22-1997.

Although normal procedures do not allow two or more cask-MCOs in close proximity in
the load-in/load-out area, this situation does not pose a criticality concern because the thick cask
shielding neutronically isolates the SNF inside each cask-MCO (HNF-SD-TP-SARP-017).

CSB procedures for loading MCOs into the standard storage tubes will require an
intermediate impact absorber be inserted between two MCOs. Analysis has shown that loading
MCOs with no intermediate impact absorber is not a criticality concern
(HNF-SD-SNF-CSER-005).

A6.4.4 Application of Double Contingency Principle

See Section 6.4.3 of the SNF Project FSAR for a general discussion about the application of
the double contingency principle. The accident evaluations presented in Sections A6.3.4.2 and
A6.3.4.3 demonstrate the application of the double contingency principle at the CSB. Two
contingencies have been identified for operation at CSB: loss of geometry control and loss of
moderator control. As discussed in Section A6.1, a loss of geometry control that results in
rubblized fuel and scrap achieving optimum size and spacing is incredible. Thus the only credible
contingency is a loss of moderator control. The factors affecting reactivity of each occurrence
have been conservatively defined, within the constraints of the design or the realistic extent of a
control loss, so that the reported consequences represent or exceed all credible situations that can
be involved in the scenario considered.

A6.5 CRITICALITY PROTECTION PROGRAM

See Section 6.5 of the SNF Project FSAR for an overview of the organizational structure
and interfaces and the technical and administrative practices of the criticality protection policy and
programs that are being developed for SNF Project operations.

In accordance with the guidance of HNF-PRO-547 the CSB is a firefighting Category B
facility. MCOs are designed to be criticality-safe when flooded by water
(HNF-SD-SNF-CSER-005), and flooding of the vault reduces neutron multiplication in the array
of MCOs stored there (see Section A6.3.4.2). Also as discussed in Section A6.3.4.2, the pre-fire
plan restricts the use of water to suppress a fire at the CSB.
A6.6 CRITICALITY INSTRUMENTATION

This section addresses the need for a criticality alarm system and a criticality detection system in the CSB. DOE Order 5480.24 references ANSI/ANS-8.3-1997, *Criticality Accident Alarm System*, for requirements relating to nuclear criticality alarm systems. ANSI/ANS-8.3-1997 states that neither a criticality alarm system nor criticality detection system is required where the probability of a criticality accident is determined to be less than $1 \times 10^{-6}$ per year.

The MCO is designed to be criticality safe even when fully moderated by water (HNF-SD-SNF-CSER-005), as it normally is until it is drained at the CVDF. Under normal conditions at the CSB, the MCOs will be dry, and without moderation the SNF stored at the CSB cannot be made critical. The receipt of a flooded MCO at the CSB was shown to be an unlikely event (SNF-4042) because of engineered features and administrative controls at the CSB and administrative controls at the CVDF.

As discussed in Section A6.3.4.3, scenarios involving misloading MCOs at the K Basin were considered in HNF-SD-SNF-CSER-005. Potential errors evaluated included loading a Mark IA fuel or scrap canister into a Mark IV basket and loading more than two scrap baskets into an MCO. No credible loading error resulted in a computed $k_{\text{eff}}$ greater than the limit of 0.95, even if the misloaded MCO was water flooded.

In all cases, with the limits and controls discussed in this chapter, the frequencies were less than $10^{-6/\text{yr}}$. Therefore, a criticality alarm system is not required at the CSB. The above discussion complies with the U.S. Nuclear Regulatory Commission equivalency requirement concerning criticality instrumentation as defined in Item 8 of HNF-4776, *Canister Storage Building Compliance Assessment, SNF Project NRC Equivalency Criteria — HNF-SD-SNF-DB-003*.

A6.7 U.S. NUCLEAR REGULATORY COMMISSION EQUIVALENCY

The requirement identified in HNF-SD-SNF-DB-003, namely that the maximum $k_{\text{eff}}$ including the contribution from any single failure be less than or equal to 0.95, has been satisfied.

A6.8 REFERENCES


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Annex A — Canister Storage Building

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Figure A6-1. Multi-Canister Overpack and Storage Tube.

CSB = Canister Storage Building.
MCO = multi-canister overpack.
Figure A6-2. Model of Multi-Canister Overpack in the Multi-Canister Overpack Handling Machine.

MCO = multi-canister overpack.
MHM = multi-canister overpack handling machine.
Figure A6-3. Model of One Multi-Canister Overpack Being Lowered into Canister Storage Building Vault.

MCO = multi-canister overpack.
Figure A6-4. Model of a Multi-Canister Overpack in the Sampling/Weld Station.

MCO = multi-canister overpack.
coolant = 50% propylene glycol and 50% water.
Figure A6-5. Model of Multi-Canister Overpacks in Vault Tubes (Finite Array).

Note that axial detail of the tubes is the same as shown in Figure A6-1.
Figure A6-6. Model of Multi-Canister Overpacks in Vault Tubes (Infinite Lattice Cell).
Figure A6-7. Neutron Multiplication Factor Versus Packing Fraction in Rubblized and Flooded Mark IA Multi-Canister Overpack with Loss of Geometry Control.
Figure A6-8. Effect of Water Between Dry Mark IV Multi-Canister Overpacks in the Canister Storage Building Vault.

Notes:

1. Error bars indicate the standard deviation of $k_{\text{eff}}$ due to statistical uncertainties in the MCNP results.

2. Results are based on the model with an infinite array of storage tubes with two Mark IV MCOs per tube. Each MCO contains three full fuel baskets (54 assemblies) and two scrap baskets. The scrap baskets have slightly different dimensions and loadings than the current design and, as a result, contain 7% more fuel mass than in the latest models.
CHAPTER A7.0

RADIATION PROTECTION
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<th>Term</th>
<th>Definition</th>
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<td>ALARA</td>
<td>as low as reasonably achievable</td>
</tr>
<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>HEPA</td>
<td>high-efficiency particulate air (filter)</td>
</tr>
<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
</tr>
<tr>
<td>MHM</td>
<td>multi-canister overpack handling machine</td>
</tr>
<tr>
<td>NRC</td>
<td>U.S. Nuclear Regulatory Commission</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
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A7.0 RADIATION PROTECTION

A7.1 INTRODUCTION

The essential features of radiation protection programs that provide for radiation exposure control, contamination control, radiological monitoring, and radiological protection instrumentation at all the Spent Nuclear Fuel (SNF) Project facilities are addressed in Chapter 7.0 of the SNF Project Final Safety Analysis Report (FSAR). Additional features of radiation protection specific to the Canister Storage Building (CSB) are addressed in this Chapter A7.0. The CSB's primary function is to provide passive interim storage of SNF in air-cooled storage tubes that isolate the SNF from the building air and surrounding atmosphere. To reach this condition, the SNF, packed inside a multi-canister overpack (MCO), is moved into the CSB in a transportation cask, placed into the cask receiving pit, moved to the sampling/weld station, and then moved into the storage tubes for long-term interim storage. Once the MCO is stored, the anticipated exposure levels of personnel depend only upon penetrating radiation resulting from efforts to monitor stored MCOs and direct radiation through the vault ceiling. Exposures to airborne radioactive material during interim storage are expected to be minimal. The major risks of receiving radiation exposure occur while unloading MCOs, staging MCOs, cap welding, and monitoring selected MCOs. The radiation protection precautions employed during MCO servicing and movement will be more extensive to protect against the possibility of the accidents described in Chapter A3.0.

A7.2 REQUIREMENTS

The requirements that form the basis for the radiation protection program are identified in Section 7.2 of the SNF Project FSAR.

A7.3 RADIATION PROTECTION PROGRAM AND ORGANIZATION

The SNF Project radiation protection program and its organization, including safety management policies and philosophies, is described in Section 7.3 of the SNF Project FSAR.

A7.4 ALARA POLICY AND PROGRAM

See Section 7.4 of the SNF Project FSAR for a discussion of the SNF Project ALARA (as low as reasonably achievable) policy and program. The SNF Project policy regarding ALARA during CSB design and construction is described in WHC-SD-W379-PMP-001, Project W-379 Canister Storage Building Project Management Plan. HNF-SD-SNF-DB-003, Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements, requires that the SNF Project incorporate, as applicable, NRC Regulatory Guide 8.8, Information Relevant to Ensuring that
Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable. The CSB radiation protection program complies with NRC Regulatory Guide 8.8, as required by U.S. Nuclear Regulatory Commission (NRC) equivalency Item 23 (HNF-SD-SNF-DB-003), using the ALARA policies noted in Section 7.4 of the SNF Project FSAR. These policies commit the CSB organization to limiting radiation exposures to ALARA levels and acknowledge that the primary method used to achieve ALARA objectives is physical design features.

During design phase ALARA analyses, a block flow diagram was developed identifying all of the individual steps required to process an MCO from receipt to long-term interim storage. This flow diagram identified an estimated time that would be required to complete each of the steps. The time estimates were then used as a basis for estimating personnel radiation exposure.

A model of the CSB processes was developed to determine the personnel exposure to radiation that could be expected during operation of the facility. This model was based on the steps in the block flow diagram. Each activity was evaluated to determine the dose rate at the location where the activity is to be performed. The workforce category and the frequency of each activity was determined from the block flow diagram, and the exposure (person-rem) for each activity was then calculated. These individual activity exposures were then summed to determine the total exposure from processing one cask-MCO. The steps that incurred higher exposures were reviewed, and design features were revised to reduce the estimated exposures.

Numerous design changes to the MCO and cask and to the CSB and its equipment have resulted from this process. Specific design features credited with reducing radiation exposure and achieving ALARA program objectives include:

- Shield wall and doors between the support building and the operating area
- Shielded operator station on the sampling crane
- Rotating shielding at the sampling/weld stations
- Temporary shields at the sampling/weld stations
- MCO shipping cask and lid
- MCO handling machine (MHM) steel shielding
- Tube plug and cover assembly
- MHM boron-infused densified wood shielding
- Service station shield hatch assembly
- Operating deck.

As the previous discussion indicates, the basic principles that were adhered to in the design of the CSB were to (1) determine the major contributors to the dose and examine methods for making the processes more efficient, and (2) provide shielding to reduce the dose.

At the CSB, the operations of primary concern from an ALARA standpoint are MCO receiving and handling operations. Cask-MCO shielding weight limitations have been imposed by the K Basins crane capacity. As a result of these shielding weight limitations, calculated
operational dose rates on the side of the MCO transportation cask are higher than desired. Also, with the transportation cask lid removed for servicing in the MCO service station, exposure levels at the top of the MCO are higher than desired. Permanent or temporary shielding has been incorporated; however, some operations may require hands-on access in radiation fields estimated at 10 to 30 mrem/h. For these operations, strict administrative controls, including personal protective equipment, will be applied to maintain exposures to ALARA levels.

A7.5 RADIOLOGICAL PROTECTION TRAINING

SNF Project requirements and criteria for radiological protection training are described in Section 7.5 of the SNF Project FSAR.

A7.6 RADIATION EXPOSURE CONTROL

SNF Project details of radiation exposure control measures are provided in Section 7.6 of the SNF Project FSAR. Details specific to the CSB follow. Items 20 and 21 of HNF-SD-SNF-DB-003 require that the SNF Project incorporate, as applicable, the following NRC requirements.


- Incorporate control devices for access to high-radiation areas that conform to the requirements of 10 CFR 20, Section 20.1601, “Control of Access to High Radiation Areas.”

For the CSB, the radiological exposure annual dose criteria of 10 CFR 72.104 have been incorporated into the CSB design and safety analysis. These criteria apply to design measures to protect any offsite public individual during normal operations and anticipated occurrences. These annual dose equivalent criteria are 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other critical organ.
For the CSB, the criteria for hourly dose limit to the public as described in 10 CFR 20.1301 have been incorporated into the design and safety analysis. This dose limit (0.002 rem) is assumed to be from external sources for any unrestricted area during normal operations and anticipated occurrences.

During the CSB design reviews, no accessible high-radiation areas (≥0.1 rem/h at 30 cm) as defined by 10 CFR 20.1601 were identified; therefore, the requirements of 10 CFR 20.1601 do not apply. If during facility operations, a high-radiation area is identified, the requirements of 10 CFR 20.1601 will apply in accordance with NRC equivalency requirements (HNF-SD-SNF-DB-003, Item 20).

Access to the radiological area of the CSB (see Figure A7-1) is controlled using one or more of the methods listed and discussed in Section 7.6.2.6 of the SNF Project FSAR. Primarily, access control will be by the use of signs and barricades at the CSB, but other measures described in Section 7.6.2.6 of the SNF Project FSAR will be used when needed.

Permanent shielding is provided in the CSB as part of the operating deck and in the design of the MHM, sampling/weld station, the MCO service station, and the biological shield wall between the operating area and the operating support area. Shielding also is provided in the operator control areas and in regions around the vaults. Some of the major shielding and special tooling that is provided is discussed below. Radiological controls are incorporated into the operating procedures for these areas.

The CSB vaults are designed as three independent areas with full-length concrete shielded walls between them. Exterior walls of the subsurface structure are designed to limit dose rates outside the vaults to 0.1 mrem/h. The vault ceilings are designed to limit dose rates above the vaults to 0.05 mrem/h. Intake and exhaust plenums are designed to avoid radiation streaming and to limit dose rates outside the plenums to a maximum of 0.1 mrem/h. The MCO storage tubes are equipped with storage tube plug assemblies constructed of concrete and steel. These plugs limit the dose rate from the MCO stored in the tube to less than 0.2 mrem/h at the surface of the operating deck. A 2-ft-thick shield wall is installed at the north end of the operating deck adjacent to the operations support area.

The MHM is designed with permanent shielding to protect from direct radiation through the sides of the MHM and from radiation streaming through interface gaps between MHM components, and between the MHM and the operating area deck surface. A retractable shield skirt is placed over the storage tube deck opening to limit radiation exposure during shield plug removal and subsequent operations in the tubes. Shielding for the MHM cask and turret system consists of a combination of steel for gamma attenuation and a borated material for neutron absorption. See Chapter A2.0 for a detailed description of MHM shielding. Overall, the MHM shielding is designed to limit gamma exposure to operations personnel to an average of 0.2 mrem/h during MCO movement and handling. Exposure rates may exceed 0.2 mrem/h for short periods during raising and lowering MCOs, but the design will ensure that averaged and
peaking exposure rates will meet ALARA principles. Neutron exposure will be negligible (<10%) with respect to the gamma dose. There will be no direct alpha or beta dose.

The service station uses the transport cask–MCO closure plug as the main shielding for operations in this area. In addition, installation of the MCO mobile station hatch assembly provides some additional shielding. With the cask, this shielding reduces the average cumulative dose rate for cask unloading operations to approximately 1.96 mrem/h. The special tooling provided for operations in this pit is the mobile containment service tent, which is a portable and high-efficiency particulate air (HEPA) filter ventilation containment, to contain any potential airborne radioactivity. The ventilation unit exhausts into the operating area. The mobile service tent is intended for use during off-normal situations to control releases and will probably never be used.

The sampling/weld station uses permanent shielding for protection of the worker from radiation. This permanent shielding consists of immovable shielding and movable, specially designed shielding blocks that allow insertion of the MCO. The permanent shielding is designed to exceed the effectiveness of the cask shielding. The shielding reduces the average cumulative dose rate during sampling or welding operations to approximately 1.86 mrem/h. In addition, a HEPA filter ventilated welding system is provided for mitigation of potential airborne radioactivity during sampling or welding operations.

Based on Title 10, Code of Federal Regulations, Part 835, “Occupational Radiation Protection” (10 CFR 835), dose rate limits, a radiation buffer zone will be established outside of the CSB in the vicinity of the load-in/load-out area for the purpose of personnel protection when a cask–MCO is located within the truck vestibule or the load-in/load-out area. Field implementation of the buffer zone will be the responsibility of the radiological control organization.

Engineering and administrative controls have been included in the CSB design to minimize airborne radioactivity. The CSB ventilation is designed to maintain approximately -0.1 in. of H₂O pressure on the operating area and consists of a building ventilation system providing approximately 44,000 ft³/min of flow divided as follows:

- Approximately 35,500 ft³/min recirculation in the operating bay
- Approximately 9,000 ft³/min HEPA-filtered building exhaust
- Approximately 4,500 ft³/min HEPA-filtered ventilation from a sampling station into the operating bay system
- Approximately 4,500 ft³/min HEPA-filtered ventilation from a weld station into the operating bay system.
These systems will be used to control any potential airborne material as close to the source as possible and to provide general ventilation of the facility. In addition to these installed systems, portable HEPA filter ventilation systems will be utilized as needed by the radiological job planning of the work. These systems range from 500 to 2,000 ft³/min and will exhaust into the operating bay.

Section 20.4 of HNF-SD-SNF-RD-001, SNF Project Standards/Requirements Identification Document, and Title 40, Code of Federal Regulations, Part 61, “National Emission Standards for Hazardous Air Pollutants” (40 CFR 61), require that doses received by the maximally exposed individual from emissions of radionuclides shall not exceed 10 mrem/yr. To meet this standard, engineered ventilation systems as described in Chapter A2.0 have been provided to abate radioactive particulate emissions. The ventilation system HEPA filters are accepted by the Washington State Department of Health as the best available radionuclide control technology. Calculations of projected emissions required by 40 CFR 61 result in the requirement for continuous airborne emission monitoring. The CSB heating, ventilation, and air conditioning airborne emissions monitoring system described in Chapter A2.0 satisfies the 40 CFR 61 monitoring system requirements. This system provides strictly sampling and monitoring functions and provides no operational control functions. Analysis results indicate that abated emissions will be less than the 10 mrem/yr maximally exposed individual limit. The CSB design prevents a release of radioactive liquid during normal operations as described in Section A2.8.6.

See the SNF Project FSAR, Section 7.6.2.4 discussion of MCO and transportation cask shielding.

A7.7 RADIOLOGICAL MONITORING

The radioactive material sampling and monitoring programs conducted internal and external to SNF Project facilities are addressed in Section 7.7 of the SNF Project FSAR. As SNF begins to arrive at the CSB, a startup surveillance and monitoring program will be implemented to define and characterize exposure rates and to ensure that neutron and gamma dose rates are in agreement with anticipated levels.

The CSB stack emissions monitoring system is described in Chapter A2.0.

A7.8 RADIOLOGICAL PROTECTION INSTRUMENTATION

A summary of the SNF Project plans and procedures governing radiation protection instrumentation is provided in Section 7.8 of the SNF Project FSAR. A description of the types and locations of radiation detection instruments used at the CSB is presented in Chapter A2.0.
A7.9 RADIOLOGICAL PROTECTION RECORD KEEPING

Radiological protection record keeping requirements are described in Section 7.9 of the SNF Project FSAR.

A7.10 OCCUPATIONAL RADIATION EXPOSURES

For the CSB, occupational radiation exposures to workers are estimated based on the following:

- Receipt and movement of MCOs into the storage vaults during the first two years of operation and the processes associated with MCO receipt and movement
- MCO validation sampling and monitoring
- Interim storage of SNF and maintenance and operations activities associated with SNF storage.

As discussed in Section A7.4, block flow diagrams and shielding analysis were used to estimate time durations to complete operational tasks, dose equivalent rates, and cumulative dose rates during unloading, sampling, welding, and handling MCOs. Transmittal FDP-788, Canister Storage Building ALARA Analysis 09, documents the ALARA assessment including assumptions and conclusions. The ALARA analysis estimates are based on assumed minimum and maximum dose rates and intervals and a selected probability distribution. For analysis purposes, it was assumed that 200 MCOs would be transported to the facility each year. The main contributors to exposure were MCO receipt and handling, and cap welding.

Occupancy within the CSB vault operation area during interim storage is estimated to be 25% of 2,000 working hours per year. Monte Carlo methods have been employed to determine the exposure levels for this occupancy. Data used to obtain the estimates using the Monte Carlo method included ranges of task duration and dose rate changes for these tasks. A detailed discussion of the Monte Carlo simulations used are included in Transmittal FDP-788.

Based on the ALARA analysis results, average operator exposure levels are expected to be as follows:

- 860 mrem/yr total effective dose equivalent for unloading, sampling, welding, and handling the MCOs
- 50 mrem/yr total effective dose equivalent for general operations during MCO storage.
Exposures for other categories of workers are considerably less. The exposure levels for handling MCOs contains an increment based on neutron dose (40 mrem/year).

A7.11 REFERENCES


NRC Regulatory Guide 8.8, 1978, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, D.C.

Figure A7-1. Radiological Control Boundary.
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CHAPTER A8.0

HAZARDOUS MATERIAL PROTECTION
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<td>ALARA</td>
<td>as low as reasonably achievable</td>
</tr>
<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
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<td>SNF</td>
<td>spent nuclear fuel</td>
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A8.0 HAZARDOUS MATERIAL PROTECTION

A8.1 INTRODUCTION

The major provisions of the occupational safety and health program, as the program applies to hazardous material protection for the Spent Nuclear Fuel (SNF) Project, are addressed in Chapter 8.0 of the SNF Project Final Safety Analysis Report (FSAR).

A8.2 REQUIREMENTS

The requirements that form the basis for the hazardous material protection program are identified in Section 8.2 of the SNF Project FSAR.

A8.3 HAZARDOUS MATERIAL PROTECTION PROGRAM AND ORGANIZATION

The SNF Project has an established, visible, and comprehensive occupational safety and health program. This program is described in Section 8.3 of the SNF Project FSAR.

A8.4 ALARA POLICY AND PROGRAMS

While there is no established formal SNF Project as-low-as-reasonably-achievable (ALARA) program for nonradiological hazardous materials, the SNF Project has expanded the classic concept of ALARA (i.e., minimization of radiological exposures) to the application of exposure minimization for all hazardous substances and conditions. The SNF Project's policy is described in Section 8.4 of the SNF Project FSAR.

Work practices for hazardous material protection and control of chemical exposures as stated in Section 8.4 of the SNF Project FSAR will be implemented at the Canister Storage Building (CSB) using approved SNF Project implementing procedures. The occupational safety and health program will use the additional provisions of Section 8.4 of the SNF Project FSAR. These will be implemented at the CSB using approved SNF Project implementing procedures.

Applicable ergonomics considerations, included in DOE Order 5480.10, Contractor Industrial Hygiene Program (Section 9B), under the occupational safety and health program, industrial hygiene subprogram, are included in Chapter A13.0. Ergonomics considerations, along with risk factor control processes and the ergonomics program, are described in the SNF Project human engineering program plan (see Section 8.4 of the SNF Project FSAR and SNF-4399, Spent Nuclear Fuel Project Human Engineering Program Plan).
A8.5 HAZARDOUS MATERIAL TRAINING

Plans and procedures for training SNF Project workers regarding hazardous materials are summarized in Section 8.5 of the SNF Project FSAR.

Section 8.5 of the SNF Project FSAR states that SNF Project management provides training, professional education, and certification opportunities necessary to support, maintain, and enhance industrial hygiene staff proficiency to meet or exceed U.S. Department of Energy industrial hygiene training objectives and goals in accordance with HNF-SD-SNF-RD-001, *SNF Project Standards/Requirements Identification Document*. CSB management is responsible for ensuring that workers assigned to any task involving hazardous materials are trained in the safety and health hazards associated with such hazardous materials. Workers will perform only those tasks for which they have received the proper training. As the CSB mission changes, CSB management will review the training requirements and modify worker training requirements accordingly. CSB management is also responsible for ensuring that retraining is provided within the time allowed by training course requirements.

A8.6 HAZARDOUS MATERIAL EXPOSURE CONTROL

Worker safety features at the CSB are an integral part of facility design and operation. The CSB design encompasses human factors considerations to ensure that operations can be conducted safely. SNF Project occupational exposures to hazardous materials and the spread of hazardous material contamination are controlled by a combination of engineered, operational, and administrative controls, and by the use of personal protective clothing and equipment. These controls are described in Section 8.6 of the SNF Project FSAR.

No significant hazardous materials have been identified at the CSB as a result of a hazard analysis that was performed and documented in HNF-SD-SNF-HIE-001, *Canister Storage Building Hazard Analysis Report*, except for the radionuclide content in the multi-canister overpacks (MCOs). Major features of worker protection are presented in Chapter A3.0 and are categorized by hazard. These features are in addition to safety-class or safety-significant features for design basis accidents. The toxicological hazards of this radionuclide inventory were evaluated in HNF-SD-SNF-TI-059, *A Discussion on the Methodology for Calculating Radiological and Toxicological Consequences for the Spent Nuclear Fuel Project at the Hanford Site*, and found to be bounded by the radiological consequences.

All work activities in the CSB will receive adequate advance planning so that if potential hazardous materials are identified due to changing conditions in the future, specific precautions will be applied. The exposure controls identified in Sections 8.6.1 through 8.6.4 of the SNF Project FSAR will then be implemented at the CSB using approved SNF Project implementing procedures.
Results of detailed CSB hazard analysis are presented in table form in HNF-SD-SNF-HIE-001. The primary inventory of hazardous material in the CSB involves the radionuclide content in the MCOs. Other hazardous material identified by the hazard identification process includes asphyxiants (diesel exhaust, helium, inert gas), biological agents (snakes, spiders), carcinogens (welding fumes), corrosives (battery acid), toxics (fuels, welding fumes, uranium, plutonium), heavy metals (lead in batteries, welding fumes), flammable or combustible materials (gasoline, diesel fuel, paint solvents), and others (used decontamination rags, glycol for cooling). No routine chemical processes will be conducted in the CSB. Some chemicals, such as those used for equipment decontamination, may be used occasionally.

Section 1.2.1 of SNF-3328, Canister Storage Building Design Basis Accident Analysis Documentation, states that there are no chemical inventories of safety concern. Table A3-1 identifies hazards by form, type, location, and total quantity.

The MCO is backfilled to a prescribed pressure at the Cold Vacuum Drying Facility, using helium, just before it is sealed for shipment to the CSB. Internal pressure can increase, as a function of time, because of the radiolytic decomposition of water and aluminum hydroxide inside the MCO and the release of hydrogen from the chemical reaction of water with metallic fuel. Uncontrolled release of this MCO internal gas pressure is an operational accident resulting from failure of MCO sampling system equipment, breach of the MCO, or operator error during the MCO sampling process.

As indicated in Chapter A4.0, safety-class and safety-significant structures, systems, and components prevent or mitigate gaseous release events. In addition, technical safety requirements are developed to govern activities associated with potentially high concentrations of hydrogen. Workers performing such activities are trained on the appropriate precautions to be taken, including evacuation of the area. The monitoring portion of the hydrogen control strategy is described in Section A8.7.

Other hazardous materials at the CSB will be properly inventoried and stored to control hazards inherent to the material in accordance with SNF Project implementing procedures.

A8.7 HAZARDOUS MATERIAL MONITORING

Summaries of the hazardous material sampling and monitoring programs that are conducted internally and externally for SNF Project facilities are provided in Section 8.7 of the SNF Project FSAR. The workplace and external monitoring program described in Sections 8.7.1 and 8.7.2 of the SNF Project FSAR will be implemented at the CSB using approved SNF Project implementing procedures. An environmental, radioactivity, and chemical emissions monitoring program, including requirements, is presented in Section 8.7.2 of the SNF Project FSAR. This program will be implemented for the CSB in accordance with approved SNF Project implementing procedures.
As discussed in Chapter A3.0, the MCO is prone to hydrogen gas generation from metal oxidation reactions and radiolysis of bound and free water. The hydrogen control strategy includes inerting of the MCOs and dilution with inert gas at the Cold Vacuum Drying Facility. As part of the process validation program, a sample of the gas in the MCO is collected and sent to a laboratory for analysis. Sample collection will be discontinued when the validation program determines sampling is no longer required.

A8.8 HAZARDOUS MATERIAL PROTECTION INSTRUMENTATION

Summaries of plans and procedures governing hazardous protection instrumentation are provided in Section 8.8 and Table 8-2 of the SNF Project FSAR. As stated in Section 8.8 of the SNF Project FSAR, safety and health specialists will determine any need for hazardous protection instrumentation and the number and placement of instruments in the CSB under normal and emergency conditions in accordance with the requirements stated in Section 8.8 of the SNF Project FSAR.

A8.9 HAZARDOUS MATERIAL PROTECTION RECORD KEEPING

The SNF Project has an established document control and records management program. This program is summarized in Section 8.9 of the SNF Project FSAR.

A8.10 HAZARD COMMUNICATION PROGRAM

The hazard communication program applies to the purchase, receipt, use, and storage of hazardous chemicals and products. This program is summarized in Section 8.10 of the SNF Project FSAR. The hazard communication program for the CSB will be implemented in accordance with provisions of Sections 8.10.1 through 8.10.6 of the SNF Project FSAR including hazard posting in work areas, chemical management, chemical labeling, chemical product list, material safety data sheets, information, and training. This will be accomplished in accordance with approved SNF Project implementing procedures.

A8.11 OCCUPATIONAL CHEMICAL EXPOSURES

Predicted annual exposures to workers from hazardous material sources are identified in Section 8.11 of the SNF Project FSAR. As discussed in Section 8.11 of the SNF Project FSAR, the projected annual exposures to workers at all SNF Project facilities, including the CSB, are expected to be negligible. Also see Section 8.10 of the SNF Project FSAR for identification of chemical hazard locations, posting, chemical management, and other controls to limit occupational chemical exposure.
A8.12 REFERENCES

1.
2. DOE Order 5480.10, Contractor Industrial Hygiene Program, U.S. Department of Energy,
   Washington, D.C.

3. HNF-SD-SNF-HIE-001, 2000, Canister Storage Building Hazard Analysis Report, Rev. 3, Fluor
   Hanford, Incorporated, Richland, Washington.

4. HNF-SD-SNF-RD-001, 1999, SNF Project Standards/Requirements Identification Document,

5. HNF-SD-SNF-TI-059, 1999, A Discussion on the Methodology for Calculating Radiological
   and Toxicological Consequences for the Spent Nuclear Fuel Project at the Hanford Site,

6. SNF-3328, 2000, Canister Storage Building Design Basis Accident Analysis Documentation,

7. SNF-4399, 1999, Spent Nuclear Fuel Project Human Engineering Program Plan, Rev. 0, Fluor
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CHAPTER A9.0

RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT
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Annex A — Canister Storage Building

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<th>Description</th>
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<td>CSB</td>
<td>Canister Storage Building</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>HEPA</td>
<td>high-efficiency particulate air (filter)</td>
</tr>
<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
</tr>
</tbody>
</table>
A9.0 RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT

A9.1 INTRODUCTION

The essential features of the radioactive and hazardous waste management programs that provide for safe control, collection, and handling of wastes generated during routine operations at Spent Nuclear Fuel (SNF) Project facilities are detailed in Chapter 9.0 of the SNF Project Final Safety Analysis Report (FSAR). This chapter only applies to waste generated within the Canister Storage Building (CSB) and those processes and systems designed to deal with that waste. The only potentially radioactive waste from normal operations at the CSB is radioactively contaminated high-efficiency particulate air (HEPA) filter elements, waste from normal radiation monitoring and decontamination processes, and waste and debris generated from the multi-canister overpack (MCO) cap welding process. Small volumes of nonradioactive hazardous and nonhazardous wastes also will be generated.

A9.2 REQUIREMENTS

The requirements that form the basis for the radioactive and hazardous waste management program are identified in Section 9.2 of the SNF Project FSAR.

A9.3 RADIOACTIVE AND HAZARDOUS WASTE MANAGEMENT PROGRAM AND ORGANIZATION

The facility administrative procedures for solid waste management contain the procedural guidance for the planning, generation, and disposal of generated waste in compliance with applicable requirements. The administrative procedures cover characterization, preplanning, designation, containerization, disposal, and programmatic requirements. A summary of the SNF Project waste management program is provided in Section 9.3 of the SNF Project FSAR.

A9.4 RADIOACTIVE AND HAZARDOUS WASTE STREAMS AND SOURCES

The following subsections identify the CSB waste streams, their sources, and their management.

A9.4.1 Waste Management Process

The goals and policies of the CSB hazardous and radiological waste management process are described in Section 9.4.1 of the SNF Project FSAR.
A9.4.2 Waste Sources and Characteristics

At the CSB, both solid and liquid wastes will be generated in small volumes and gas emissions will be released. The expected generation rates of each CSB waste stream are listed in Table A9-1. The following subsections describe the sources of solid and liquid waste summarized in Table A9-1.

Table A9-1. Waste Generation Rates for the Canister Storage Building.

<table>
<thead>
<tr>
<th>Waste stream</th>
<th>Canister Storage Building</th>
</tr>
</thead>
<tbody>
<tr>
<td>Solid</td>
<td></td>
</tr>
<tr>
<td></td>
<td>HEPA filter elements = 0.9 m³/yr (32 ft³/yr), assuming eight units changed every year (potentially radioactive)</td>
</tr>
<tr>
<td></td>
<td>Rags, wipes, solidified decontamination solutions, anticontamination tapes, protective clothing, vegetation, carcasses, and welding debris &lt; 1 m³/yr (35 ft³/yr) (potentially radioactive)</td>
</tr>
<tr>
<td></td>
<td>Fluorescent light ballasts &lt; 0.2 m³/yr (7 ft³/yr) (nonhazardous)</td>
</tr>
<tr>
<td></td>
<td>Fluorescent light bulbs &lt; 2 L/yr (0.5 gal/yr) (recycled)</td>
</tr>
<tr>
<td></td>
<td>Waste oils &lt; 19 L/yr (5 gal/yr) (potentially hazardous)</td>
</tr>
<tr>
<td></td>
<td>Batteries &lt; 50 L/yr (13 gal/yr) (replacement of two to three crane batteries per year [recycled])</td>
</tr>
<tr>
<td>Liquid</td>
<td>Cooling coil condensate &lt; 150 L/yr (40 gal/yr) (nonhazardous)</td>
</tr>
<tr>
<td>Airborne</td>
<td>There is the potential for airborne emissions to be generated that are radioactive gases and particulate. These have been estimated, and the resulting offsite dose is within Title 40, Code of Federal Regulations, Part 61, &quot;National Emission Standards for Hazardous Air Pollutants&quot; (40 CFR 61), standards. The emission control and monitoring systems satisfy state and federal requirements and are permitted.</td>
</tr>
</tbody>
</table>

HEPA = high-efficiency particulate air (filter).

A9.4.2.1 Solid Waste Streams and Sources. The HEPA filters will be handled as radioactive solid waste as warranted by the level of activity present. At the CSB, loaded HEPA filters will be generated as a solid waste or a potentially radioactive waste. HEPA filter elements will be used for the MCO servicing system, MCO handling machine, and MCO sampling hood. Chapter A2.0 discusses the operating circumstance under which these filter elements can receive a loading of radioactive particulate. The filters are not expected to have hazardous materials deposited on them. Other process HEPA filters serve as a backup for the MCO filters and should not be contaminated. The building heating, ventilation, and air conditioning HEPA filters are a third level of backup and also are not expected to be contaminated. Solid waste debris is generated during normal operations, equipment maintenance, monitoring of stored SNF, and during any necessary cleanup following abnormal events. Radioactive wastes will be generated from decontamination necessary for maintenance and repair of contaminated equipment, such as the tube vent and purge carts or MCO service station enclosure. The debris wastes will consist of...
contaminated protective clothing, small tools, equipment, vegetation, animal carcasses, and welding debris.

A9.4.2.2 Liquid Effluents and Sources. The only expected water discharges at the CSB are sewage from the temporary office building (a trailer located near the CSB), which will be tied to a septic tank and drain field. A small amount of condensate from the CSB heating, ventilation, and air conditioning cooling coils (less than 40 gal/yr) will be generated during the more humid months of June and July.

A9.4.2.3 Gaseous Effluents and Sources. The CSB is permitted for airborne emissions by DOE/RL-98-30, Radioactive Air Emissions Notice of Construction Canister Storage Building (Revised Sealing Configurations for Spent Nuclear Fuel) Project W-379. See Section 9.4.2.3 of the SNF Project FSAR for general information regarding gaseous effluents and sources. See Chapters A2.0 and A7.0 for information regarding CSB airborne emission criteria and monitoring.

A9.4.3 Waste Handling or Treatment Systems

Solid radioactive wastes including heating, ventilation, and air conditioning HEPA filters; HEPA filters from MCO handling; and other operational wastes (e.g., rags and wipes) will be packaged per the Waste Management Federal Services of Hanford, Incorporated waste acceptance requirements defined in HNF-EP-0063, Hanford Site Solid Waste Acceptance Criteria, described in Section 16 of HNF-SD-SNF-RD-001, SNF Project Standards/Requirements Identification Document, and in facility waste management plans, and then transported to Waste Management Federal Services of Hanford, Incorporated, for disposition.

Hazardous materials such as batteries and fluorescent light bulbs will be picked up and recycled by commercial vendors or other appropriate site disposal methods. Hazardous waste such as waste oils will be collected in approved collection areas and shipped for disposal per site procedures and per the requirements of DOE Order 5820.2A, Radioactive Waste Management.

Sewage from the temporary office building will be sent to a septic tank and drain field. The condensate from the CSB heating, ventilation, and air conditioning cooling coils will be discharged to the ground consistent with the requirements of ST 4509, Washington State Department of Ecology Waste Water Discharge Permit.

Other potential nonregulated wastes will be handled as described in Section 9.4.2.1.3 of the SNF Project FSAR.

Airborne wastes are filtered by HEPA filters and then released per permit as noted in Section A9.4.2.3.
There is no special treatment system for fire sprinkler water in the case of actuation. Water will be noncontaminated and will be allowed to run out of the doors and other building openings.

A9.5 REFERENCES


CHAPTER A10.0

INITIAL TESTING, IN-SERVICE SURVEILLANCE,
AND MAINTENANCE
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<td>CSB</td>
<td>Canister Storage Building</td>
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<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
</tr>
<tr>
<td>MHM</td>
<td>multi-canister overpack handling machine</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
</tr>
<tr>
<td>SSC</td>
<td>structure, system, and component</td>
</tr>
</tbody>
</table>
A10.0 INITIAL TESTING, IN-SERVICE SURVEILLANCE, AND MAINTENANCE

A10.1 INTRODUCTION

Essential features of the initial testing program, the operational readiness review, the in-service surveillance program, and the maintenance program implemented at Spent Nuclear Fuel (SNF) Project facilities are described in Chapter 10.0 of the SNF Project Final Safety Analysis Report (FSAR). Canister Storage Building (CSB)-specific features of these programs are described in this Chapter A10.0.

A10.2 REQUIREMENTS

The requirements that form the basis for the initial testing program, the operational readiness review program, the in-service surveillance program, and the maintenance program are identified in Section 10.2 of the SNF Project FSAR.

A10.3 INITIAL TESTING

The SNF Project initial testing program ensures the operability of equipment and facilities before facility operation. Project details of this program are provided in Section 10.3 of the SNF Project FSAR.

A10.4 IN-SERVICE SURVEILLANCE PROGRAM

The SNF Project in-service surveillance program is designed to maintain the integrity of facility systems and to ensure that systems perform their function of protecting the health and safety of the public, workers, and facility staff by prevention or mitigation of accident consequences. Details of this program are provided in Section 10.4 of the SNF Project FSAR. The technical safety requirements that include the surveillance requirements for the CSB are described in Chapter A5.0.

A10.5 MAINTENANCE PROGRAM

Section 10.5 of the SNF Project FSAR summarizes the maintenance program that supports safe operation of the SNF Project facilities. The structures, systems, and components (SSCs) that are important in the mitigation and prevention of analyzed accidents and are included in the formal CSB maintenance program are identified in Chapters A3.0 and A4.0 of this document.
The maintenance program for the SNF Project facilities is conducted in accordance with DOE Order 4330.4B, *Maintenance Management Program*, which provides general policy and objectives for establishing cost-effective maintenance and repair of U.S. Department of Energy property. The maintenance program will incorporate the results of considerable project activity and subsystem vendor interface activity aimed at ensuring an acceptable design and acceptable operating practices relative to reliability, availability, and maintainability.

Operational design information has been used as input to a process system design basis capacity study of the SNF Project documented in HNF-SD-SNF-RPT-011, *Spent Nuclear Fuel Project Design Basis Capacity Study*. This study used a summary-level model of major SNF Project facilities to determine the impact of facility interactions on the overall time to complete fuel removal operations. The results indicate that to support a CSB throughput of one multi-canister overpack (MCO) per day, one sampling/weld station is required, whereas two sampling/weld stations are provided in the CSB design. Two stations permit sampling and welding operations to proceed concurrently.

HNF-S-0425, *Specification for Spent Nuclear Fuel Canister Storage Building*, includes maintainability and availability requirements for the facility and specifies a 40-year service life for the facility structures and systems that directly support interim storage and a 5- or 10-year life for SSCs used in the preparation of MCOs for storage (e.g., the sampling/weld station). HNF-S-0425 specifies the following concepts for use in the CSB design:

- Maximize equipment interchangeability
- Operate power transmission devices below 75% of ratings
- Provide adequate equipment materials for the operating environment
- Utilize commercially available equipment
- Identify equipment repair methods and egress routes
- Locate complex components including electronic devices or those having a high probability of failure in nonradioactive areas
- Cranes and hoists required for handling radioactive waste shall meet designated requirements including those in the applicable portions of ASME NOG-1-1995, *Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)*.

Operations, startup, and maintenance personnel have participated in the design process through support of design reviews and participation in appropriate portions of factory testing of SSCs, specifically the MCO handling machine (MHM), the tube vent and purge cart, and sampling/weld station equipment. These review activities have resulted in suggested
improvements regarding operability and maintainability. These proposed improvements have been presented to the design organization and evaluated in subsequent design activities to identify impacts on capital, operations, and maintenance costs. Design reviews to date have identified no high-maintenance components.

The design organization has included in subcomponent purchase requests (by applying Specification 01730, Operations and Maintenance Data [W-379-C-CSB-01730]) requirements that vendors provide maintenance recommendations for their components and subsystems based on past experience. These recommendations include spare parts inventories, maintenance and repair procedures and suggested frequencies, and requirements for special maintenance tools. Most of the CSB equipment designs apply commercially available components, and vendors are required to include maintenance accessibility and testability in their designs.

Early in the CSB Project, a reliability analysis plan was developed to guide the preparation of the assessment of the availability and total operational efficiency of the facility. The plan was attached to Transmittal FRT-2660/9600454, Reliability Analysis Plan (Jacobs 1996). Its purpose was to identify cost-effective improvement measures in the design. Several failure modes and effects analyses were performed both on material handling equipment and on key CSB process SSCs including the receiving crane, the vault, the deck, the tube vent and purge cart, and the MHM. Although the emphasis of these analyses was on safety, productivity and maintainability issues were identified as appropriate. Results of these analyses were used to influence designs to enhance the reliability and maintainability of CSB systems and components.

The Project recognizes the critical role of the MHM in the overall success of the CSB facility. Accordingly, numerous studies and analyses have been performed to enhance the reliability, operability, and maintainability of this essential system. Failure modes and effects analyses for two subsystems (control system and hoist and grapple) have been performed. In addition, the analysis documented in Letter DESH-9760688, Reliability/Availability/Maintainability Analysis for the Multi-Canister Overpack Handling Machine (Daughtridge 1997), was performed for the entire MHM based on the 90% design status point. This analysis concluded that the probability of the MHM operating for 8 hours without failure was 98.1%, and identified an availability of the MHM of 90%. The mean times to repair were determined to be relatively short (on the order of a few hours). The analysis in Letter DESH-9760688 (Daughtridge 1997) points out that further simplification of the MHM control system design to reduce the number of interlocks would achieve significant reliability and availability improvements. Subsequently, an independent review of the MHM design was conducted primarily for the purpose of simplifying the design. At about the same time, a second television camera was included in the design for positioning the MHM. A mockup of the control panel was constructed, and a human factors review was conducted. Results of the human factors review were assessed by operations personnel, and an action plan for grouping and labeling controls was adopted. The results of all of these activities coupled with a change in the CSB technical basis (sealed MCO strategy) have resulted in a simpler, more reliable and dependable MHM.
In summary, considerable interface has occurred between organizations contributing to the
design of the CSB relative to availability, operability, and maintainability, and the results have
been considered in the present CSB and MHM designs. This interchange of information will
continue as the facility progresses into startup testing and operational phases.

The SNF Project has completed failure mode and effects analyses and fault tree analyses as
part of the FSAR hazards assessments, which includes evaluating routine maintenance activities.
There are no maintenance activities identified other than the specific technical safety requirement
surveillance items required for operations to be bounded by the authorization basis. The SSCs are
designed to place the MCO in a safe and stable state without operator action. Preventive
maintenance and calibrations are utilized to provide continued operability of facility systems and
components.

The maintenance plan to be described in the SNF maintenance implementation plan, which
is submitted to the U.S. Department of Energy, Richland Operations Office for approval, will
implement a graded approach to CSB maintenance activities based on sound engineering
judgment and knowledge of the facility and will take into consideration the following items.

- Results of the reliability, availability, and maintainability analysis performed on the
  MHM.
- With the exception of the MHM, the design life for the equipment used during the
  preparation for storage is a minimum of 5 years, whereas the expected operational
  life of these systems is 3 years. This short operating life would argue against an
  extensive predictive maintenance program and would justify a simple maintenance
  history program.
- The passive nature of the systems and components supporting interim storage also
  would seem to justify a simple preventive maintenance and overhaul program.
- Wherever practicable, the CSB equipment designs incorporate commercially available
  equipment with readily available spare parts and a proven record of performance.
- No components are subjected to severe duty conditions.
- Availability of an MHM maintenance pit to facilitate the periodic servicing
  recommended by the vendor, plus MCO and tube plug grapple surveillance and load
  testing.
- The sampling/weld stations have built-in redundancy.

Policies and procedures are in place to effectively manage SNF Project facility maintenance
activities. The SNF Project FSAR, Section 10.5, summarizes the maintenance policies and
procedures that are implemented at the CSB and also describes the process used during the
development of the detailed content of the maintenance plan. The SSCs that are important in the
mitigation and prevention of analyzed accidents and that are included in the CSB maintenance
plan are identified in Chapters A3.0 and A4.0. Following completion of the MCO
loading/welding operations, the SNF Project will define the details of the CSB MHM layup
maintenance activities based on operating experience and vendor recommendations.

A10.6 REFERENCES

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CHAPTER A11.0

OPERATIONAL SAFETY
Annex A — Canister Storage Building

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<th>Description</th>
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<tbody>
<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>NRC</td>
<td>U.S. Nuclear Regulatory Commission</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
</tr>
</tbody>
</table>
A11.0 OPERATIONAL SAFETY

A11.1 INTRODUCTION

Features of the Spent Nuclear Fuel (SNF) Project conduct of operations program and fire protection programs are described in the following sections and in Chapter 11.0 of the SNF Project Final Safety Analysis Report (FSAR).

A11.2 REQUIREMENTS

The requirements that establish the basis for conduct of operations and general aspects of operational safety are identified in Section 11.2 of the SNF Project FSAR.

A11.3 CONDUCT OF OPERATIONS

"Conduct of operations" is a set of principles that establishes an overall philosophy for achieving excellence in the operation of the SNF Project facilities. SNF Project application of conduct of operations principles is described in Section 11.3 of the SNF Project FSAR.

A11.4 FIRE PROTECTION

The SNF Project facilities fire protection programs are addressed in Section 11.4 of the SNF Project FSAR. The results of a Canister Storage Building (CSB) fire hazard analysis, documented in HNF-SD-SNF-FHA-002, Final Fire Hazard Analysis for the Canister Storage Building, are addressed in Chapter A3.0 and are summarized in the following subsections.

A11.4.1 Fire Hazards

The fire hazard analysis documented in HNF-SD-SNF-FHA-002 was developed in accordance with Section 12.4.6 of HNF-SD-SNF-RD-001, SNF Project Standards/Requirements Identification Document. The analysis comprehensively assessed the risk from fire at the CSB to determine the following: (1) that the potential for occurrence of fire is minimized; (2) that requirements that will provide an acceptable degree of life safety to SNF Project and contractor workers are in place, and there are no undue hazards to the public from fire and its effects in SNF Project facilities; (3) that the CSB facilities achieve “nuclear safety equivalency” to comparable U.S. Nuclear Regulatory Commission (NRC)-licensed facilities; (4) that the Uniform Building Code (ICBO 1994) Type II-N building construction criteria for the CSB is adequate as compared to the NRC recommended Type I building construction criteria; (5) that the fire protection requirements in Title 10, Code of Federal Regulations, Part 72, “Licensing Requirements for the
Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste,”
Section 72.122(e) “Proximity of Sights” (10 CFR 72), are general and comparable to the
requirements in DOE Order 5480.7A, Fire Protection, and DOE Order 6430.1A, General Design
Criteria; and (6) that safety systems are not damaged by fire. The fire hazard analysis along with
the accident analyses determined that a fire would not cause onsite or offsite release of
radiological or other hazardous materials that would threaten the public health and safety or the
environment. The fire hazard analysis states that consideration should be given to following the
evolving designs of the new SNF Project Path Forward facilities to reconfirm that the CSB fire
protection features continue to provide adequate fire protection meeting the requirements of DOE
Order 5480.7A, and DOE Order 6430.1A.

The CSB fire hazard analysis (HNF-SD-SNF-FHA-002) was prepared to meet the
requirements of DOE Order 5480.7A. As required, this analysis addressed the following
elements. No unacceptable conditions were identified.

- Description of construction
- Protection of essential safety-class equipment
- Fire protection features
- Description of fire hazards
- Life safety considerations
- Critical process equipment
- High value property
- Damage potential; maximum credible fire loss and maximum possible fire loss
- Fire department or brigade response
- Recovery potential
- Potential for toxic, biological, and/or radiation incident due to a fire
- Emergency planning
- Security considerations related to fire protection
- Natural hazards (earthquake, flood, wind) impact on fire safety
- Exposure fire potential, including the potential for fire spread between fire areas.

Facility modifications will be screened to fire hazards analysis requirements and will receive
an unreviewed safety question review.

The CSB consists of two adjoining fire areas: the operating area (which includes the three
storage vault areas, the operating deck, and the trailer vestibule), and the operations support area
(which includes the control room, the change room, and electrical and mechanical equipment
rooms). Analysis of the consequences and probabilities of a worst-case fire for the CSB indicates
that the maximum possible fire loss for the facility would be realized in a fire in the operations
support area. An operating area fire was also evaluated, but losses from a fire event were
determined to be bounded by an operations support area fire because of limited and dispersed
combustible loading within the operating area. The total cost of facility cleanup and equipment
and structure replacement of an operations support area fire would be $11.72 million.
This amount does not include any environmental cleanup of the radionuclides dispersed from the
CSB high-efficiency particulate air filtration system that is included in the postulated fire zone. Because the multi-canister overpacks vent through local high-efficiency particulate air filters at the sampling/weld station before exhausting through the building filters, the radionuclide content on those filters is anticipated to be relatively low. No toxicological or biological consequences resulting from any credible fire are anticipated. Refer to HNF-SD-SNF-FHA-002 for additional detail. The maximum possible fire loss for the operating area is $5.18 million resulting from a fire involving hydraulic fluid and lubricating oil on the multi-canister overpack handling machine trolley drive assembly.

A11.4.2 Fire Protection Program and Organization

The fire protection program for the SNF Project facilities is structured and implemented in accordance with the operating contractor's safety management policies, philosophies, and the criteria identified in SNF Project administrative procedures. CSB-specific aspects of the fire protection program are provided in the following paragraphs.

The operating area and operations support area are separated by two-hour fire rated walls, and the operations support area is protected by a wet pipe sprinkler system. The design also includes automatic shutdown of appropriate heating, ventilation, and air conditioning units upon receipt of a fire alarm. The normal water supply to the facility fire protection systems is the service water system. A second water supply is provided by a storage tank and pumping system dedicated to the fire water loop around the CSB and Building 2704-HV. The CSB fire protection system is classified as general service.

DOE Order 5480.7A and the Uniform Building Code (ICBO 1994) invoked by DOE Order 6430.1A require sprinkler protection for the operating area. An exemption from the DOE Order 5480.7A requirement and a deviation from the DOE Order 6430.1A and Uniform Building Code (ICBO 1994) requirements have been approved by U.S. Department of Energy, Richland Operations Office, as documented in Letter 96-SFD-320, Contract No. DE-AC06-96RL13200 — Project W-379, Spent Nuclear Fuel Canister Storage Building (CSB) Review and Approval of Exemption and Deviation Requests from Automatic Fire Sprinkler/Fire Suppression System Requirements (Sellers 1996). The basis for the deviation and exemption is that sprinklers provide little value to either prevention or mitigation of a hydrogen fire or the other fires evaluated in the fire hazard analysis.

Life safety criteria are met as required by NFPA 101, Life Safety Code. Life safety in the facility is satisfactory.
A11.4.3 Combustible Loading Control

The absence of a sprinkler system in the operating area places an even greater importance on the control (quantity and location) of combustibles in this area; thus, the operating deck will not be used as an area for storing fixed or transient combustible materials.

Calculations contained in HNF-SD-SNF-FHA-002 provide limits for the amount and location of combustibles to preclude damage to the building structure and high-value equipment. Administrative controls will be used to control combustible quantities and separation distances based on HNF-SD-SNF-FHA-002, Table 5-4, calculations. Per HNF-SD-SNF-FHA-002, the following administrative controls will be employed to control combustibles within the operating area with respect to defined fuel packages, arrangement, separations from other fuel packages, and locations on the operating deck.

- Combustible materials inherent to all equipment proposed for use within the operating area shall be as low as reasonably achievable. This includes limiting the use of materials with combustible components and, whenever possible, replacing them with either noncombustible or fire resistant or retardant components.

- To limit the damage potential resulting from a vehicle fuel spill and fire in the trailer vestibule, the receiving crane shall be restricted by administrative procedure or control from that area during those periods when a truck or tractor is located within the trailer vestibule. In addition, a dam in the receiving area crane rail trench prevents diesel fuel from flowing into the load-in/load-out and operating areas.

- Transient combustible materials shall be limited to the minimum type and quantities required to perform the transient activity or operation and shall be removed upon completion of that activity or operation.

- The potential fire size resulting from a transient combustible fire shall be controlled by restricting the size and quantity of potentially combustible materials and maintaining the minimum separation distances identified in HNF-SD-SNF-FHA-002, Table 5-4.

- The leakage of fluid from the receiving crane or multi-canister overpack handling machine shall require the immediate safe shutdown of operations until corrective action to seal the leak is completed and the fire hazard inherent to the leak is eliminated.

These controls will be implemented through operating procedures as prescribed in HNF-SD-SNF-RD-001, Section 12.2.1.
A11.4.4 Fire Fighting Capabilities

The Hanford Fire Department maintains a training program for fire fighting, fire testing, and fire inspection. Fire fighting capabilities that apply to all SNF Project facilities are addressed in Section 11.4.4 of the SNF Project FSAR. CSB-specific fire response procedures are addressed in the following paragraph.

All areas of the CSB are automatically monitored for fire and smoke. Fire alarm features include transmission of signals to the Hanford Fire Department via a radio fire alarm reporter. A fire brigade is not required at the CSB due to the close proximity of the Hanford Fire Department. The standard response to an alarm condition in the 200 East Area is by the Hanford Fire Department from the 200 Area Fire Station, which is fully staffed, trained, and equipped. Hanford Fire Department response time to the CSB is approximately five minutes. These are the response times and the responder locations assumed in HNF-SD-SNF-FHA-002. When manned, a crew from the 100 Area Fire Station also is dispatched, with an estimated response time of 10 minutes. Vehicle access to the CSB is provided by a paved access road. Refer to WHC-SP-1180, Hanford Site Emergency Response Needs, Volume I, “Needs Assessment,” and Volume 2, “Master Plan,” for additional description of Hanford Fire Department response capabilities.

A11.4.5 Fire Fighting Readiness Assurance

A prefire plan for the CSB will be prepared by the Hanford Fire Department prior to facility operations. HNF-SD-SNF-FHA-002 provides a description of fire fighting methods that will be employed in the plan.

Due to the limitation on using large quantities of water in the operating area, fire fighting methods will include, as a first line approach, the following:

- Portable fire extinguishers
- Low-expansion foam
- Water fog.

If these methods fail to extinguish a fire, water hose streams may be employed.

A summary of SNF Project fire prevention inspections, fire safety drills and exercises, and program record keeping requirements is provided in Section 11.4.5 of the SNF Project FSAR.

A11.5 REFERENCES


CHAPTER A12.0

PROCEDURES AND TRAINING
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LIST OF TERMS

FSAR  final safety analysis report
MHM  multi-canister overpack handling machine
SNF  spent nuclear fuel
A12.0 PROCEDURES AND TRAINING

A12.1 INTRODUCTION

A description of the essential features of Spent Nuclear Fuel (SNF) Project procedures and training programs is provided in Chapter 12.0 of the SNF Project Final Safety Analysis Report (FSAR).

A12.2 REQUIREMENTS

The requirements that form the basis for the SNF Project procedures and training programs are identified in Section 12.2 of the SNF Project FSAR.

A12.3 PROCEDURE PROGRAM

SNF Project activities are conducted in accordance with written procedures. A summary of the facility procedures program, including development and maintenance of procedures, is provided in Section 12.3 of the SNF Project FSAR and its subsections.

A12.4 TRAINING PROGRAM

The objective of the SNF Project personnel training program is to provide and maintain a qualified work force for safe and efficient facility operations. A summary of the SNF Project personnel training program—including training development, maintenance of training, and modification of training materials—is provided in Section 12.4 of the SNF Project FSAR and its subsections.

Training needs associated with the multi-canister overpack handling machine (MHM) are determined and developed using a systematic approach to training as stated in Section 12.4.1 of the SNF Project FSAR. A job task analysis that identifies the knowledge and skills required to operate the MHM has been performed. Operations personnel working with the MHM will be required to obtain a certification based on the knowledge and skills identified in the job task analysis. Certification will require passing written examinations and performance evaluations. For the certification of the initial operating crews, it has been determined that knowledge-based training will be primarily presented in the classroom and skills-based tasks will be primarily addressed by on-the-job training activities. Training emphasis will be placed on on-the-job training and will include operation of the MHM with dummy multi-canister overpacks as part of the equipment check dry runs or as part of special training evolutions. Consistent with the
SNF Project training program, detailed MHM operator training will include the following subjects:

- MHM systems including interlocks, controls, and indications
- MHM operations procedures
- MHM alarms and alarm response procedures
- MHM accidents, malfunctions, and emergency procedures.

Instructor qualification will be completed as stated in Section 12.4 of the SNF Project FSAR. MHM operating experience is included in the continuing training program as determined by line management.

Maintenance personnel providing support to the MHM will receive appropriate classroom and on-the-job training as determined by the job task analysis.
CHAPTER A13.0

HUMAN FACTORS
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<tr>
<th>Abbreviation</th>
<th>Definition</th>
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<tbody>
<tr>
<td>ALARA</td>
<td>as low as reasonably achievable</td>
</tr>
<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>HEPA</td>
<td>high-efficiency particulate air (filter)</td>
</tr>
<tr>
<td>HFE</td>
<td>human factors engineering</td>
</tr>
<tr>
<td>HMI</td>
<td>human-machine interface</td>
</tr>
<tr>
<td>HVAC</td>
<td>heating, ventilation, and air conditioning</td>
</tr>
<tr>
<td>MCO</td>
<td>multi-canister overpack</td>
</tr>
<tr>
<td>MHM</td>
<td>multi-canister overpack handling machine</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
</tr>
<tr>
<td>SSC</td>
<td>structure, system, and component</td>
</tr>
</tbody>
</table>

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A13.0 HUMAN FACTORS

A13.1 INTRODUCTION

Features of the Spent Nuclear Fuel (SNF) Project human factors engineering (HFE) program are described in Chapter 13.0 of the SNF Project Final Safety Analysis Report (FSAR). Specific application of the human factors process to the Canister Storage Building (CSB) is described in this chapter.

The human factors process applies to all operations, facilities, personnel, programs, procedures, and staff associated with the CSB. Human factors were considered through system development (planning, requirements analysis, system design, and system testing and evaluation) as evidenced by the CSB performance specifications documented in HNF-S-0425, Specification for Spent Nuclear Fuel Canister Storage Building, and by procurement specifications. Several HFE analyses of significant CSB structures, systems, and components (SSCs) and operations were also completed during the design phase. The results from these analyses were used to guide the design and procedure development of those SSCs.

The formality and the extent of the systematic inquiry into the human factors associated with the CSB were determined on the basis of the extent of the human interaction, the system design effort, and the risk associated with human performance failures over the life cycle of the CSB project. For example, the most concentrated human activity occurs when the multi-canister overpacks (MCOs) are being accepted and moved into storage. Human activity then declines to that of a monitoring role with periodic inspections and minimum maintenance activity.

The welding equipment, tube vent and purge cart, and the sampling system incorporate appropriate human factors considerations. The tube vent and purge cart vendor design specifications (at paragraph 2.1.7.4. Human Factors Design) include appropriate human factors performance design criteria. These are discussed in this chapter. All systems for the CSB have been reviewed for human–machine interfaces (HMIs). For the sampling/weld station, frog tool, and the tube vent and purge cart, the reviews have been performed on design documentation only. Further HMI review of the equipment will be completed during the factory acceptance testing. The SNF Project will evaluate and report on the final design and implementation in subsequent updates to this FSAR.

This chapter describes the results of the HFE and ergonomics analysis completed on the existing equipment; the review of administrative issues impacting human performance at the CSB (e.g., procedures, staffing, and training); a review of the existing design specifications and documents for SSCs; and, finally, a review of human actions as initiators and mitigators identified in HNF-SD-SNF-HIE-001, Canister Storage Building Hazard Analysis (see Section A3.3).
A13.2 REQUIREMENTS

The requirements that establish the basis for HFE are identified in Section 13.2 of the SNF Project FSAR.

A13.3 HUMAN FACTORS PROCESS

See Section 13.3 of the SNF Project FSAR for a description of the human factors process that was applied to the CSB design. A lessons-learned process is in place for CSB engineers. Lessons learned and operating experience are disseminated as defined in HNF-PRO-067, *Managing Lessons Learned*. Information regarding lessons learned is routinely distributed by the SNF Project Lessons Learned point of contact and acknowledged by the appropriate engineers. The SNF Project has established points of contact whose job is to coordinate the distribution of lessons learned to the appropriate organizations or individuals within the SNF Project. The point of contact typically adds lessons-learned information to required reading, or if applicable, distributes it to the procedures group, training group, operations organization, or maintenance organization. If applicable, lessons learned then are implemented by the organization or group (i.e., procedure modifications, upgraded training, or revised maintenance practices) to avoid recurrence of an adverse work practice or operating experience. The lessons-learned program serves to validate that human tasks can be adequately performed. If a task is found to be outside of human capabilities and limitations, the lessons-learned program is set up to establish corrective action to fix the discrepancy.

A13.4 IDENTIFICATION OF HUMAN-MACHINE INTERFACES

This section summarizes the safety-class and safety-significant SSCs that require HMI to function. The significant HMIs for general-service SSCs pertinent to CSB operation are listed.

A systems requirements analysis using a graded approach was performed as an integral part of the CSB design process. A number of human factors considerations were included in the CSB design, in accordance with HNF-S-0425, *Human System Interface Design Review Guidelines*, and NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, Section 18.1, "Human Factors Engineering, Control Room," were reviewed in addition to U.S. Department of Energy guidelines. Application of these guidelines to the CSB for HMIs have been documented for the safety-related SSCs in SNF-3907, *Spent Nuclear Fuel Project Canister Storage Building Human Factors Engineering (HFE) Analysis: Results and Findings*.

The systems listed in Table A13-1 were reviewed. A supplemental HFE and ergonomics review of the CSB has been performed. Table A13-1 shows the application of the graded approach and consideration of HMI in that review. Some systems will continue to be followed as those systems' designs are nearing completion. The heating, ventilation, and air conditioning
Table A13-1. Summary Analysis for Human Factors Engineering and Ergonomics Graded Approach.

<table>
<thead>
<tr>
<th>CSB structures, systems, and components important to safety*</th>
<th>Significant HMIb</th>
<th>Selected for HFE and ergonomic analysis using graded approachc</th>
</tr>
</thead>
<tbody>
<tr>
<td>Support area building</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>Receiving crane positioning and interlock control system, receiving crane lifting yoke</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>Rail frogs</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>MCO and transportation cask</td>
<td>No</td>
<td>Yes</td>
</tr>
<tr>
<td>Mobile service station tent gantry hoist</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>MCO flex connector, HEPA filter, and piping between the filter and cask</td>
<td>No</td>
<td>Yes</td>
</tr>
<tr>
<td>MCO servicing instruments</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>MCO tube vent and purge cart</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>MHM closed-circuit television system</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>MHM operations control room</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>Remaining components of MHM and the collision avoidance system</td>
<td>No</td>
<td>Yes</td>
</tr>
<tr>
<td>Sampling/weld station</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>MCO sampling equipment (sample hood and HEPA filter, exhaust system [HEPA filter], sample hood exhaust flow indicator), cooling cap</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>Sampling/weld station test equipment (welding equipment and weld diagnostics equipment, sampling cart and tools, including long-reach tools)</td>
<td>Yes</td>
<td>Yes</td>
</tr>
<tr>
<td>Distributed control system</td>
<td>No</td>
<td>Yes</td>
</tr>
<tr>
<td>Heating, ventilation, and air conditioning system</td>
<td>No</td>
<td>Yes</td>
</tr>
<tr>
<td>Airborne emissions monitoring system (gaseous effluent monitoring system)</td>
<td>No</td>
<td>Yes</td>
</tr>
<tr>
<td>Communication system</td>
<td>Yes</td>
<td>Yes</td>
</tr>
</tbody>
</table>

Notes:

*System safety classification is provided in Chapter A2.0.

bSignificant HMI includes either operation of system or maintenance of system, or both operation and maintenance of system.

cA system may not have significant HMI but be selected for evaluation of limited HMI following system installation.

CSB = Canister Storage Building.
HEPA = high-efficiency particulate air (filter).
HFE = human factors engineering.
HMI = human–machine interface.
MCO = multi-canister overpack.
MHM = multi-canister overpack handling machine.
(HVAC) system has been reviewed using specification drawings, interviews with the HVAC
engineer and architect-engineer, and by observation at the CSB and has been found to meet or
exceed HFE accessibility requirements. General-service systems at the CSB usually do not
include significant HMI; however, accessibility for routine periodic maintenance by the CSB
operators appears to be acceptable. The general-service systems' accessibility is confirmed during
pre-operational testing and evaluation when the operators will “dry run” all procedures. Much of
the maintenance interaction, in accordance with vendor instruction, will include periodic
changeout of high-efficiency particulate air (HEPA) filters, instrument readings, and calibration
and minor adjustments to the system where necessary.

The systems with “yes” in the Table A13-1 “Significant HMI” column were analyzed using
the HFE checklists, interviews with engineers, observation, or review of design specifications and
drawings. Additional analysis was completed on the following systems or program elements using
established HFE criteria recognized by the U.S. Department of Energy and referenced in the CSB
performance specification document (HNF-S-0425):

- Multi-canister overpack handling machine (MHM)
- Instrumentation and controls
- Operations and maintenance
- Normal and abnormal operations (including recovery actions)
- Maintenance activities
- Procedure development
- Staffing.

Documents that contain the established HFE criteria recognized by the U.S. Department of
Energy include MIL-STD-1472, Human Engineering Design Criteria for Military Systems,
Equipment, and Facilities; and UCRL-15673, Human Factors Design Guidelines for
Maintainability of Department of Energy Nuclear Facilities.

A13.5 OPTIMIZATION OF HUMAN–MACHINE INTERFACES

A13.5.1 Results of the Human Factors Engineering
Analysis of Normal Operations

See Section 13.5.1 of the SNF Project FSAR for a description of the HFE checklist
process. Normal human operations at the CSB are characterized by the flowchart in
Figure A13-1. Operations are involved in determining the appropriate sequence of operational
steps and tools used. Operational block sequence diagrams have been completed for CSB
operations, and the block sequence diagrams have been further detailed into minute-by-minute
activity at the operator level. This forms the basis for a detailed task analysis and critical task
analysis, and provides the training organization and the procedure writers a wealth of information
from which appropriate detailed procedures and training can be constructed. The CSB startup
plan includes a mock-up exercise of MCO handling activities, processing, testing, and other
operations for verification and validation of normal and abnormal operations.

A few items of equipment or tools were undergoing engineering or re-engineering at the
time of the HFE evaluation. An assessment of the HFE work accomplished shows a very high
degree of consideration for human–machine interfacing. For example, where possible, machines
are doing the work of lifting and carrying items of significant weight. The CSB will be operating
with a minimum crew.

It is clear from the review of the SSCs in the beginning of this chapter that HMI is minimal,
with regard to the facility as a whole. The graded approach method presented at the beginning
also indicated that many systems will not have human interaction other than possibly routine or
periodic maintenance. Outside of the major items of equipment (e.g., the receiving crane and
MHM) much of human activity will be involved with sampling, recording data, routine
maintenance activities as described in the system design descriptions, security, communications,
and housekeeping. HMI occurs in a number of critical areas, the most notable of which is
operator control of the MHM or receiving crane. The CSB process is not a fast process, such as
an assembly line, but a slow deliberate process, with lengthy process time between the major
actions.

Where appropriate, designs include HFE inputs or administrative features have been
implemented to meet the intent of the HFE checklists. For example, if there is a biomechanical
disadvantage created in the design, because of ALARA (as low as reasonably achievable)
requirements, an engineering change will first be sought to overcome the disadvantage, taking
into account factors such as cost of change, and time required to redesign based on project
schedule. Engineering is keyed to the practice that when the engineering change is not feasible
because of project factors, administrative controls will be put into place (e.g., certification of
operator capability to perform the task, training before performance of the task).

A radio-controlled crane allows the operator to control the crane from a distance and not
have to use the panel controls; it also allows the operator to better observe the crane operation
and work with the grappling mechanism that attaches to the cask during receiving operations.
The crane control serves as a backup to radio control operations.

The receiving cask lid is removed from the transportation cask by using an automatic bolt
removal and torquing device.

The control room in the support building incorporates adjustable furnishings. The
environmental design (e.g., illumination, HVAC, noise, and aesthetics) meets the requirements of
DOE Order 6430.1A, General Design Criteria, Section 1300.

The HVAC, gaseous effluent monitoring system, and instrument air and service air systems
need mainly routine maintenance performed on them or are interfaced with infrequently.
Attention was focused on HEPA filter accessibility for changeouts, instrument accessibility for
calibrations, and ease of removal and replacement of failed components. The components are readily accessible. Maintainability is the human engineering key to these systems.

The weld pit has been designed with awareness of the biomechanical forces on the human when using long-reach tools. The design concentrates on making the task as simple and as quick as possible to reduce the risk factors associated with the task.

The tube vent and purge cart has included appropriate human factors requirements in the vendor specification. These human factors requirements will be checked during acceptance testing.

The CSB HFE analysis (SNF-3907) includes a checklist evaluation for each CSB subsystem or component, as appropriate. Included are identified deficiencies that were brought to the attention of engineering, and where appropriate, solutions that were suggested. If no identified deficiencies are cited, the system meets the HFE criteria or the checklist criteria are not applicable. The identified deficiencies have been checked and evaluated for appropriate resolution concurrent with pre-operational testing and startup of operations.

The following discussion describes the various human-machine-intensive equipment and systems in the CSB and the impact of the CSB on the operator. The operating ambient temperature (60 °F to 85 °F) is acceptable. The acoustic environment, vibrational environment on the MHM and the receiving crane, and the available illumination at the floor and equipment panels are to be measured by the facility staff during testing. The quality and quantity of the available light are to be evaluated during startup testing and task lighting provided where required. Human performance is to be tested under operating conditions with special attention to the concerns about vibration. The engineering changes and administrative controls identified during the HFE reviews discussed above will be verified and validated during operational testing.

A13.5.1.1 Multi-Canister Overpack Handling Machine. Human factors reviews were completed on the MHM. These were documented in SNF-3907, which concluded that the HFE was satisfactory based upon correction of several items. Letter FDH-9855462, Update of 22 Issues from the U.S. Department of Energy, Richland Operations Office Review of Multi-Canister Overpack Handling Machine, Supporting Key Driver Resolution Committee Process for Multi-Canister Overpack Handling Machine Installation (Williams 1998), stated, “A human factors review was performed as part of the DE&S Hanford, Inc. review of the Multi-Canister Overpack Handling Machine (MHM) control console.” The letter further stated that “significant changes” had been made to the control console layout and color coding. The supplementary HFE and ergonomics evaluations of the MHM were documented in the SNF-4831, Human Factors Engineering and Ergonomics Analysis for the Canister Storage Building: Results and Findings (Supplemental Report), which concluded that enhancements had been made to correct previously identified deficiencies.

A13.5.1.2 Receiving Crane. The receiving crane is designed to move a cask with a loaded MCO from the transporter in the trailer vestibule to the cask receiving pit and then to move a cask
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with an empty MCO onto the transporter at the trailer vestibule. Review of the receiving crane layout against the appropriate human factors criteria referenced in the CSB performance specification (HNF-S-0425) resulted in the following human factors concerns:

- Joystick controls can be inadvertently activated
- Main and auxiliary hoist controls on the control panel appear inconsistent with direction of operation.

Inadequacies identified during the human factors review of the control panel layout of the crane have been appropriately resolved using engineering and administrative controls.

The cask yoke may be handled by two methods: (1) an additional observer with radio connection to the crane operator manually guides the cask yoke to the proper cask-yoke connecting features, or (2) the crane operator can be positioned on the trailer and, using remote control, manually guide the yoke to the proper cask-yoke connecting features.

A13.5.1.3 Tube Vent and Purge Cart. The tube vent and purge cart system is designed to monitor and maintain an inert gas environment around any MCO placed into the overpack storage tubes as well as to monitor the atmosphere in the tubes. The cart includes features to analyze gas from the storage tube, purge generated gases (containing hydrogen) through a HEPA filter, and replace the purged gas with an inert gas.

The cart contains piping and instrumentation necessary to remove air from an overpack storage tube and replace it with inert gas. The system consists of inert gas supply, flexible steel hoses, sampling connection, gas monitor loop with heat exchanger, continuous air monitor unit, radioactive gas monitor, hydrogen gas monitor, associated interlocks and alarms, HEPA filter, and the oxygen monitor and associated alarms. Each of the alarms is annunciated and monitored locally. The operations include tube pressure purging, tube refilling, and tube venting.

The operations sequence to set up the tube vent and purge cart was included as part of the CSB human factors review. During any abnormal or accident operations, whether directly associated with the tube vent and purge operations or elsewhere in the facility, the cart is placed into safe configuration. The design basis has been defined, and a piping and instrumentation diagram is available and was reviewed. CSB project plans to design this and other equipment to the criteria included in the CSB performance specifications. The criteria include appropriate human factors criteria. W-379-P-1305-1305, Tube Vent and Purge Cart Assembly, the tube vent and purge cart specification, calls out appropriate human factors performance criteria. Vendor submittals include HFE requirements. The HFE review concluded that the cart adequately incorporates HFE criteria.

A13.5.1.4 Instrumentation and Controls. The CSB project design requirements for instrumentation and controls include requirements for equipment used in the process, HVAC, and radiation monitoring systems. The requirements exclude hand-held portable monitoring.
equipment. The basic philosophy is to use standard, commercial, off-the-shelf equipment consistent with DOE Order 6430.1A. All mobile equipment are to be monitored and controlled locally, including the tube vent and purge cart. All design media use the English unit system and colors are mostly manufacturer’s standard. CSB performance specifications (HNF-S-0425), which include appropriate human factors criteria, govern the design of any needed instrumentation and control equipment. Therefore, the instrumentation and controls were considered appropriately designed.

A13.5.1.5 Distributed Control System. The distributed control system is designed to provide personnel with status of various process systems and alarms. The system does not provide control functions for the operators. During operations, the distributed control system is not expected to be continuously manned.

During the human factors review of this system, several performance enhancement suggestions were provided to the CSB Project management. Each of these enhancements was considered by the CSB management.

The supplemental HFE and ergonomic review found that the distributed control system demonstration screen displays, in some instances, did not provide adequate contrast between background color and foreground color lettering. Maximum contrast between background and foreground colors is preferred, so what needs to be legible can, in fact, be seen. Engineering will ensure adequate screen contrast is achieved.

A13.5.1.6 Environmental Considerations. CSB lighting systems comply with the Lighting Handbook (IES 1987) and ASHRAE 90-75R, Energy Conservation in New Building Design. The lighting systems include exterior, interior, standby, and emergency lighting. Exterior lighting illuminates the outside of the buildings using wall-mounted, low-pressure, sodium light fixtures and is controlled by light-sensitive photocells in compliance with DOE Order 5480.1A, Environmental Protection, Safety, and Health Protection Standards, Section 1650-1. Interior lighting consists of normal and standby lighting. Illumination levels follow DOE Order 6430.1A, which recommends 50 footcandles for workstation lighting, 30 footcandles for work area lighting, and 10 footcandles for non-work area lighting. Fluorescent light fixtures with energy-saving ballasts are used in the support building rooms. High-intensity discharge lighting fixtures are used in the operating area shelter. Standby lighting systems are controlled using manual light switches. Emergency lighting consists of individual lighting fixtures powered by an integral battery and self-luminous exit lighting that comply with NFPA 101, Life Safety Code, and DOE Order 6430.1A, Section 1635-1. The emergency lighting fixtures are installed in areas essential to life safety—ingress and egress routes to and from the equipment areas. Portable lighting may be needed during inspection of the cask surface to meet the requirements of ANSI/ANS-57.2-1992, Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants, which calls for 100 footcandles for inspection at the surface of the cask.

Acoustical features in the support area control room include a suspended acoustical tile ceiling and sound board, which are installed on separate metal stud-framed wall partitions on both...
sides of the wall adjoining the empty generator room. The acoustical features are provided to comply with the requirements of DOE Order 6430.1A, Section 950.

The HVAC system is designed to provide, along with physical barriers, part of the contamination confinement system and contamination control within the CSB. The HVAC system provides a controlled pressure gradient flow of air from outside the CSB inward through uncontaminated areas to potentially contaminated areas of the building and out through HEPA filters and a monitored exhaust. The HVAC system also provides climate control to ensure that environmental conditions in the CSB are maintained in the required ranges for personnel and equipment.

Operational requirements for the support area HVAC subsystem are as follows: a support area temperature of 72 °F to 78°F and an entry-related room operating pressure of -0.03 in. water gauge. Maintenance and surveillance requirements for the support area HVAC subsystem include the following.

- Air handlers CSB-AH-003 and CSB-AH-004 filters will be routinely changed.
- Condensing unit CSB-CU-003 will be periodically inspected and cleaned.
- Support area HVAC components and conditions will undergo periodic surveillance.

A13.5.1.7 Mobile Service Station Tent and Hatch Assembly. The mobile service station tent is undergoing redesign and should be put into an HFE and ergonomics tracking system to follow the redesign effort.

A13.5.1.8 Rail Frogs. Specialy designed tools are required to lift, turn, and set back down the rail frogs. The design of the rail frog tools appropriately took into account HFE and ergonomics.

A13.5.1.9 Sampling/Weld Station. The primary issues at the sampling/weld station are ALARA concerns, thus the use of a glove box and special manual extension tools in the sampling/weld station pit to install and operate an MCO valve and a tube hookup to the MCO, which is then hooked up to the sample cart. The design calls for two gantries with hood and ventilation requirements.

A13.5.1.10 Communications System. The design of the CSB communications system provides the cabling and raceway system equipment needed to connect the telephone, public address, intercom, and radio communications systems within the CSB to the communications equipment interface point. All communications system physical interfaces are located in the communications area of the uninterruptible power supply room. A door from the electrical equipment room provides access to the uninterruptible power supply room. Space is provided for telephone exchange equipment, paging equipment, wire terminating frames, radio equipment, and coaxial and fiber-optic transmission equipment.

Telephone requirements are based on providing internal and external communications service to all office and operating areas within the CSB. The telephone system provides tie-ins to
the Hanford Site crash alarm and the Hanford Site 911 emergency telephone systems. An
intercom system is incorporated into the telephone system.

Public address system coverage is required for all areas accessible to personnel, both
interior and exterior. Public address systems within the CSB are accessed through specified
telephone sets within the facility. Master overrides originate from the control room and the
emergency response center. The evacuation alarm, complying with DOE Order 6430.1A,
Section 1300-6.5.5, will be annunciated through the public address system speakers. The
evacuation alarm can be activated from the control room or the emergency response center.

Area alarms are part of the 200 East alarm system, and CSB alarms are incorporated into
the distributed control system. Rimgdown phones are installed for security use. Fire alarm and
detection system signals are transmitted to the central fire alarm system by means of a radio
system integral to the fire alarm control panel. Radiation and hazardous conditions alarms are
annunciated by means of the distributed control system. The Hanford Site crash alarm is
annunciated through the general CSB telephone system ringer and is directed to selected CSB
telephones.

A13.5.2 Staffing

A13.5.2.1 Operator Capabilities. See Section 13.5.2.1 of the SNF Project FSAR for a
discussion on how human factors are addressed for CSB operator capabilities.

A13.5.2.2 Staffing Plan. See Section 13.5.2.2 of the SNF Project FSAR for a discussion of the
SNF Project staffing plan. The CSB staff is responsible for all facility operations in the CSB,
including the following:

- Staging, storing, and monitoring of MCOs
- Controlling locks and tags
- Controlling facility access
- Providing contamination and radiation control
- Handling laundry
- Operating systems
- Controlling radiation area access
- Maintaining radiological control routines
- Supporting radiological work area control as needed
- Supporting seal welding of MCOs.

Two normal 8-hour shifts are planned, with provision for a third shift if necessary. The
activity job hazard/safety analysis and prejob meetings provide the operators with information to
help them identify and control or mitigate hazards. CSB administrative procedures require job
hazard analyses as part of the job planning. Monthly, quarterly, and yearly safety inspections are
conducted to identify unsafe conditions throughout the facility, including the CSB equipment.
Unsafe conditions result in postings, personnel protective equipment requirements, and timely corrective actions, as appropriate. Any technical safety requirements also provide for the appropriate staffing with a basis and a qualification requirement. Chapter A5.0 provides the necessary technical safety requirements information. Chapter A17.0 includes provisions for staffing that may be updated as the operations are finalized. Changes to the staffing plan will be reviewed to ensure that appropriate human performance has been considered (i.e., knowledge, skills, abilities, availability, and sufficient depth). This approach results in human factors being appropriately considered in the staffing plan.

A13.5.3 Procedures

See Section 13.5.3 of the SNF Project FSAR for a discussion of how the human factors process applies to CSB procedures. Procedures are the principal job performance aid used at the CSB and direct the way in which the majority of human tasks are performed. Considering human factors in the development and maintenance of procedures ensures that the procedure is easy to read and follow, the steps of the procedures are presented in a logical and correct order, all documentation (such as labels, equipment representations, and diagrams) contained in the procedure is accurate, and tasks required by the procedure are within the operator’s capabilities.

The documents and standards that form the basis for the procedures and training program are found in HNF-SD-SNF-RD-001, Spent Nuclear Fuel Project Standards/Requirements Identification Report.

A13.5.4 Training

See Section 13.5.4 of the SNF Project FSAR for a discussion on how human factors are addressed for CSB personnel training. The purpose of CSB training is to ensure that all staff have the knowledge, skills, and abilities required to safely operate and maintain CSB facilities, equipment, and processes. Performance-based training techniques are used for operator and maintainer training to ensure that the operators and the maintainers working with the CSB acquire the skills appropriate for successful task performance through initial training and maintain those skills through periodic retraining.

Human factors issues requiring training may be identified during job and task analyses (the CSB training organization currently has a file with over 500 pages of task analyses for all CSB operating systems). Input to the analyses comes from a determination of required skill levels, current skill levels, task complexity, maintenance requirements, and lessons learned. The identified human factors issues are considered and appropriate responses implemented where applicable during the design and development of training. A direct link between the training department and the procedure development staff is provided during the development of procedures to verify that content development includes the necessary training for new procedures or operations.
From an institutional safety perspective, a stable organizational environment and management chain is necessary to ensure that CSB can attain and maintain proficiency required for safe operation. Stability of the organization ensures that links between training, operations and maintenance, and procedures group are maintained to ensure the quality and accuracy of training and operating and maintenance procedures with respect to facility configuration.

Chapter 17.0 of the SNF Project FSAR illustrates the links between the different organizations. Further details of the CSB training program and HFE aspects within the training program are described in Chapter 12.0 of the SNF Project FSAR. Chapter 12.0 encompasses engineering training, operator certification, maintenance qualification, instructor qualification, development of course content, training documentation, and relationship to other programs.

**A13.5.5 Maintenance**

See Section 13.5.5 of the SNF Project FSAR for a discussion of how the human factors process applies to the CSB maintenance program. The work management process in HNF-PRO-069, *Maintenance Management*, describes how work activities that impact facility operations can be safely and effectively managed. The implementation of design changes (i.e., construction and installation) after engineering has completed the design work is a subset of these work activities. The work management process involves several steps. Planning checklists are provided to serve as guidelines for the planning phases of work packages from pre-work planning through pre-operational or post-installation testing and operations acceptance. HFE is addressed in the planning checklist of the work management process. For example, modification packages are checked to ensure training and procedure changes impacted by the modification are adequately addressed. At the completion of construction work for complex modifications and before turnover to operations, a readiness assessment is performed to ensure the modified SSC is safe to operate and within the approved Authorization Basis. Items evaluated during the readiness assessment process typically include completion of all procedures and training documentation. Work package information is checked to ensure that special skills or knowledge required for task performance are used when identified. The work management system is further described in Chapter 10.0 of the SNF Project FSAR.

The provisions for the initial testing, in-service surveillance, and maintenance programs are applied, as appropriate, to the CSB equipment. Three types of testing will be accomplished before startup of the CSB: factory acceptance testing, construction acceptance testing, and pre-operational acceptance testing. The factory acceptance tests demonstrated to the satisfaction of the applicable design authority that the designed equipment can perform its intended operational functions. The major pieces of designed equipment include the MHM, receiving crane, tube vent and purge cart, sampling/weld station equipment, and rail frogs handling tools. The construction acceptance tests demonstrate that the installation matches the design and that all equipment is functional. The pre-operational acceptance tests demonstrate that the design and installation are operable and the equipment can perform its intended functions.
A13.5.6 Abnormal Operations (Including Recovery Actions)

Human actions during abnormal or accident operations have been reviewed by the appropriate CSB personnel. Several recovery action flowcharts were developed to cover each credible postulated incident. Flowcharts are designed to guide recovery team actions during an abnormal or accident operation. The flowcharts were reviewed for human factors concerns, and the action processes were revised to resolve human factors issues.

A procedure involving recovery from a dropped cask–MCO provides the necessary steps to respond to a dropped MCO or a dropped cask containing an MCO in the CSB and includes a checklist to aid in the development of a recovery plan to recover from an event. The initial steps are to stop the activity associated with the dropped item, ensure that equipment is in a safe configuration, evacuate personnel from the operations area to a staging area, initiate appropriate first aid as necessary, establish boundaries around the operating area, and notify the appropriate managers or other emergency personnel. Later, a recovery team will be assembled to develop an appropriate recovery plan. Once the plan is approved, it will be initiated. These completed technical procedures will be covered in detail during the operator training.

The CSB hazard analysis report (HNF-SD-SNF-HIE-001) was also reviewed to isolate those scenarios involving human error as a cause. The facility hazard analysis review ensures that those postulated events include mitigators sufficient to reduce the probability of human error. The CSB hazard analysis (HNF-SD-SNF-HIE-001) covered normal, intended, CSB operations for handling and storing a sealed MCO. Also identified and analyzed were the potential hazards associated with storing an off-normal MCO in an overpack storage tube following undetermined accident recovery actions. The following normal CSB operations were considered:

- Receiving the transporter containing the cask–MCO and moving it into the facility
- Moving the cask–MCO to the service area and removing the cask lid
- Transporting the MCO from the service area to the storage tube with the MHM
- Transporting the MCO from the storage tube to the MCO sampling/weld station and returning it to the storage tube after sampling
- Conducting activities during MCO staging and interim storage.

The following off-normal MCO storage operations were also considered:

- Event or accident leading to MCO damage has been terminated and recovery actions completed
- Off-normal MCO is in place in the overpack storage tube
Overpack storage tube plug cover is installed

An inert atmosphere has been established in the overpack tube.

The hazard identification process systematically and comprehensively identified hazards that can contribute to the uncontrolled release of radioactive or hazardous materials or that can threaten the safety of facility workers. The hazard evaluation process identified hazardous conditions, determined causes and preventive and mitigative features, and qualitatively estimated the frequency and consequences of occurrence (HNF-SD-SNF-HIE-001).

Section 7.2 of SNF-3907 discusses those hazards associated with human error. The report contains a summary of information from the hazard analysis (HNF-SD-SNF-HIE-001), and covers the following seven CSB areas:

- Truck vestibule (TV)
- Service area (SA)
- Operating area, including overpack storage tubes and tube vent and purge cart (OA)
- Sampling/weld station (WS)
- Vault (VL)
- Support building (SB)
- Outside (OU).

The HFE and ergonomics review of the CSB hazard analysis report (HNF-SD-SNF-HIE-001) did not result in a list of human initiators that were not accounted for by the human factors design review. It also did not disclose any human factors initiators that had a high frequency based upon the human initiator. Each of the scenarios involving human initiators included a set of mitigators the hazard analysis team developed. The recommended mitigators were evaluated, and those considered necessary to ensure safe operation were implemented. The results of this review were considered adequate from a human factors standpoint.

A13.5.7 Conclusions

Interviews conducted with engineers and the architect-engineer, design authorities, procedure writers and trainers; use of selected checklists to capture the requirements of DOE Order 6430.1A and guidelines in NUREG 0700 and NUREG 0800; review of specification and drawing; and direct observation of the CSB facility all demonstrate that human factors and ergonomics have been appropriately and adequately considered in the design of the CSB. The
application of human factors continue into the pre-operational testing and evaluation of the entire CSB facility, where each step that the operators and maintainers are expected to accomplish are observed for any deficiencies or unsafe human performance.

A13.6 REFERENCES


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Figure A13-1. Normal Human Operations at the Canister Storage Building.
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CHAPTER A14.0

QUALITY ASSURANCE
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<td>CSB</td>
<td>Canister Storage Building</td>
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<td>FSAR</td>
<td>final safety analysis report</td>
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<td>MCO</td>
<td>multi-canister overpack</td>
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<td>NRC</td>
<td>U.S. Nuclear Regulatory Commission</td>
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<td>OCRWM</td>
<td>Office of Civilian Radioactive Waste Management</td>
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<td>QARD</td>
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A14.0 QUALITY ASSURANCE

A14.1 INTRODUCTION

An introduction to the quality assurance program that includes the objectives and scope that apply to all Spent Nuclear Fuel (SNF) Project quality assurance activities is provided in Chapter 14.0 of the SNF Project Final Safety Analysis Report (FSAR).

A14.2 REQUIREMENTS

The requirements that form the basis of the quality assurance program are identified in Section 14.2 of the SNF Project FSAR. Additional requirements for the Canister Storage Building (CSB) are identified in the following paragraphs. These requirements ensure the quality assurance requirements for federal repository acceptance of SNF and U.S. Nuclear Regulatory Commission (NRC) equivalency requirements are satisfied.

The U.S. Department of Energy, Richland Operations Office (RL), has directed in Letter 95-SFD-098, DOE/RW-0333P (Sellers 1995), that DOE/RW-0333P, Quality Assurance Requirements and Description, published by the Office of Civilian Radioactive Waste Management (OCRWM), be applied as the principal quality assurance document to the SNF Project OCRWM program. RL has directed application of OCRWM Quality Assurance Requirements and Description (QARD) document (DOE/RW-0333P) to the following SNF Project-related activities:

- Characterization or data collection for input and use
- Conditioning into final form
- Handling, packaging, and transportation.

Items, activities, and documentation determined to be important to safety are presented in Table 3-1 in HNF-SD-SNF-RPT-007, Application of the Office of Civilian Radioactive Waste Management Quality Assurance Requirements to the Hanford Spent Nuclear Fuel Project. This document identifies certain high-level structures, systems, components, activities, and documentation that require application of DOE/RW-0333P requirements to ensure compliance with RL direction.

HNF-SD-SNF-RPT-007, Appendix E, identifies specific items, activities, and documentation that require application of QARD requirements, including the following:

- Systems
  - The sample cart features used to provide controlled refill of monitored multi-canister overpacks (MCOs) with helium
The laboratory instruments used to ensure cleanliness and helium quality (e.g., helium gas purity analyzer, an instrument periodically used to ensure that non-approved materials are not introduced into the monitored MCOs)

The helium supply system that connects to the sample cart and passes from there through the sample hood to the MCO

The helium sampling procedure and the approved analytical laboratory procedure used to determine helium purity will comply with OCRWM QARD requirements. The methods of QARD application for the items associated with the quality of the helium supply to the monitored MCO is defined in HNF-SD-SNF-RPT-007, Appendix E, Section E.5

Components

- Helium sample container
- Helium purity analyzer
- Monitored MCO gas sample container
- MCO gas sample analyzer
- PIT-721, pressure indicator transmitter
- TIT-723, temperature indicator transmitter

Activities and documentation

- Records of helium received and the sampling and analysis of helium to ensure non-approved materials are not admitted to a monitored MCO; calibration records of the instruments used to provide these data
- Data developed by CSB that are also needed by OCRWM as identified in formal direction from RL
- Qualification records involving personnel who refill monitored MCOs with helium
- Data collected and controlled during the CSB operations to show compliance with the QARD requirements and summarized and inserted into each MCO data package (traveler) upon completion of periodic monitoring activities
Procedure for operation of sampling station

Procedure for analyzing CSB helium samples

Procedure for analyzing CSB monitored MCO gas samples

Procedure for calculating amount of hydrogen present in a monitored MCO based on sample temperature, pressure, and hydrogen content

Analysis instrument calibration procedure

Qualification of personnel.

The U.S. Department of Energy has established a regulatory policy (Grumbly 1995) that new SNF Project facilities achieve nuclear safety equivalency with NRC-licensed facilities. An evaluation, documented in HNF-SD-SNF-DB-002, *Spent Nuclear Fuel Project Path Forward, Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities*, identified requirements to establish nuclear safety equivalency that are to be met in addition to existing and applicable U.S. Department of Energy requirements. These requirements, except those related to the design basis earthquake, are contained in HNF-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward Additional NRC Requirements*. WHC-SD-SNF-DB-004, *Spent Nuclear Fuel Project Seismic Design Criteria — Nuclear Regulatory Commission Equivalency Evaluation Report*, contains the design basis earthquake requirements.

The NRC nuclear safety equivalency requirements identified in HNF-SD-SNF-DB-003 that will be applied to the CSB include:

- Incorporation of requirements into safety-class procurement specifications that require suppliers to report defects and noncompliances in items or services (Item 15)
- Review and approval by RL of changes to HNF-MP-599, *Project Hanford Quality Assurance Program Description*, that could be interpreted as decreasing the quality assurance program's existing commitments for the CSB (Item 16)
- Implementation of the Project Hanford Management Contract occurrence reporting system for the design and construction of the CSB (Item 17)
- Ensuring that the appropriate quality requirements in existing Project Hanford Management Contract procedures and instructions identified in HNF-SD-SNF-DB-002 remain in effect (Item 18) (a detailed evaluation documented in Attachment A of HNF-SD-SNF-DB-002 indicates that Project Hanford Management Contract procedures and instructions are equivalent to the requirements of Title 10, *Code of Federal Regulations*, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste")
(10 CFR 72), Subpart G, "Quality Assurance," and of Title 10, Code of Federal
Regulations, Part 50, "Domestic Licensing of Production and Utilization Facilities"
(10 CFR 50), Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and
Fuel Reprocessing Plants")

- Institution of a process to identify safety-class equipment that has been identified in
the commercial nuclear power industry as being potentially defective (Item 19)

- Ensuring that the areas of vendor and subcontractor quality assurance records and
control of purchased material, equipment, and services considered important to safety
receive emphasis during CSB audits, surveillances, and assessments (Item 19).

Additional requirements apply to an MCO when it resides in the CSB. These requirements
are identified in HNF-SD-SNF-DB-005, Spent Nuclear Fuel Project Multi-Canister Overpack
Additional NRC Requirements.

Design requirements for natural phenomena hazards, other than seismic design
requirements, are identified in WHC-SD-SNF-DB-009, Canister Storage Building Natural
Phenomena Hazards.

The documents cited in this chapter identify the requirements to achieve nuclear safety
equivalency with NRC-licensed facilities and to meet the requirements of DOE/RW-0333P.
SNF-4948, Spent Nuclear Fuel Project Quality Assurance Program Plan, provides for
implementation of these requirements. A graded approach will be used for items and activities
important to safety (i.e., safety-class, safety-significant, and certain general-service items and
activities) in accordance with HNF-MP-599 and NUREG/CR-6407, Classification of
Transportation Packaging and Dry Spent Fuel Storage System Components According to
Importance to Safety.

A14.3 QUALITY ASSURANCE PROGRAM ORGANIZATION

A summary of the SNF Project quality assurance program, including summaries of safety
management policies and philosophies used as a basis for the program, is provided in Section 14.3
of the SNF Project FSAR.

The SNF Project organizational structure, responsibilities, authorities, and interfaces that
apply to the CSB are addressed in Chapter 17.0 of the SNF Project FSAR.
A14.4 QUALITY IMPROVEMENT

Descriptions of SNF Project management programs and processes used to correct adverse conditions affecting quality at all SNF Project facilities are provided in Section 14.4 of the SNF Project FSAR.

A14.5 DOCUMENTS AND RECORDS

A description of the SNF Project document control and records management program associated with quality assurance is provided in Section 14.5 of the SNF Project FSAR.

A14.6 QUALITY ASSURANCE PERFORMANCE

An overview of the SNF Project process to ensure that the performed work meets requirements is provided in Section 14.6 of the SNF Project FSAR and its subsections. The subsections address work processes, design activities, the procurement process, program tests and inspections, and independent assessments.

The CSB incorporates portions of the design and the completed below-grade sections of the canceled Hanford Waste Vitrification Plant. All of the designs and completed sections of the Hanford Waste Vitrification Plant were reviewed and met or exceeded the requirements of the SNF Project CSB. All of the below-grade structures are safety class. Portions of the vault that were completed prior to suspension of the Hanford Waste Vitrification Plant project in 1993, including the base slab and portions of the walls, were reviewed, approved, accepted, and incorporated into the present design by the architect-engineer.

A14.7 REFERENCES


HNF-3553 REV 0

Annex A — Canister Storage Building


CHAPTER A15.0

EMERGENCY PREPAREDNESS PROGRAM
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<th>Acronym</th>
<th>Description</th>
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</thead>
<tbody>
<tr>
<td>BED</td>
<td>building emergency director</td>
</tr>
<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
</tr>
<tr>
<td>DOE</td>
<td>U.S. Department of Energy</td>
</tr>
<tr>
<td>EOC</td>
<td>Emergency Operations Center</td>
</tr>
<tr>
<td>EPZ</td>
<td>emergency planning zone</td>
</tr>
<tr>
<td>ERO</td>
<td>emergency response organization</td>
</tr>
<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
</tr>
<tr>
<td>SNF</td>
<td>spent nuclear fuel</td>
</tr>
</tbody>
</table>
A15.0 EMERGENCY PREPAREDNESS PROGRAM

A15.1 INTRODUCTION

A description of the generic philosophy, objectives, and organization of the Spent Nuclear Fuel (SNF) Project emergency preparedness program for response to emergencies at the SNF Project facilities is provided in Chapter 15.0 of the SNF Project Final Safety Analysis Report (FSAR). This Annex A chapter presents emergency management information specific to the Canister Storage Building (CSB).

A15.2 REQUIREMENTS

The requirements that form the basis for the SNF Project emergency preparedness program are identified in Section 15.2 of the SNF Project FSAR.

A15.3 SCOPE OF EMERGENCY PREPAREDNESS

Potential CSB emergencies could span the spectrum of identified emergencies for SNF Project facilities, from worker injuries to general emergencies with potential public impact. The spectrum of emergencies that the CSB emergency preparedness program is designed to encompass is described in Section 15.3 of the SNF Project FSAR and in Chapter A3.0.

A15.4 EMERGENCY PREPAREDNESS PLANNING

SNF Project emergency preparedness planning includes identification of emergency organizations, assessment actions, notification processes, emergency facilities and equipment, protective actions, training, drills, exercises, and recovery actions. A summary of the emergency response organization (ER0) that is activated during CSB emergencies is provided in Section 15.4 of the SNF Project FSAR and its subsections.

A15.4.1 Emergency Response Organization

Section 15.4.1 of the SNF Project FSAR presents information related to the organizational structure and support resources to meet emergency event requirements. This information also applies to the CSB. The Hanford Site ERO has the responsibility to take actions in response to a CSB emergency to prevent or minimize impacts to workers, the public, facilities, and the environment. Initial directions and control of emergency response at the CSB before establishment of the Incident Command Organization, as described below, is the responsibility of the CSB ERO.
The Hanford Site ERO, responding to a CSB emergency, will consist of two components: the Incident Command Organization and the U.S. Department of Energy (DOE) Hanford Emergency Operations Center (EOC), as stated in Section 15.4.1.1 of the SNF Project FSAR.

**A15.4.1.1 Incident Command Organization.** The Incident Command Organization will consist of the CSB ERO and Site contractor emergency personnel (i.e., Hanford Fire Department and Hanford Patrol). The Incident Commander is the senior Hanford Fire Department official; the Hanford Patrol will provide the Incident Commander for security emergencies. The Incident Commander will direct all emergency response efforts at the CSB and has overall responsibility for the health and safety of all personnel at the event scene.

The CSB ERO, including the building emergency director (BED), will become part of a consolidated Incident Command Organization and functions under the direction of the Incident Commander. In this role, the BED retains responsibility for CSB operations and direct configuration control over CSB systems and components. The Incident Commander controls the overall management strategy associated with the emergency event and ensures that all functional areas are staffed and working effectively to mitigate the accident.

The Incident Command Organization has the authority to commit all SNF Project resources (equipment and personnel) in response to any emergency and to request supporting resources. Other responsibilities include the following: (1) implementing the CSB's building emergency plan; (2) ensuring that the CSB ERO is fully staffed and trained; (3) initially assessing, categorizing, and classifying events; (4) notifying the Patrol Operations Center and applicable contractor and DOE management through the Occurrence Notification Center; (5) implementing protective actions; (6) establishing an initial CSB Incident Command Post; (7) controlling the event scene; (8) initiating mitigating activities; and (9) initiating recovery actions when directed. At the CSB, the BED is a certified operations shift manager.

A listing of the primary and alternate BEDs by title, work location, and work telephone numbers is contained within the CSB's building emergency plan. The BED is on the CSB premises during hazardous operations and is available through an "on-call" list 24 hours a day at all other times. Operations maintains a listing of on-call BED names, with work and home telephone numbers, at the Occurrence Notification Center.

The Incident Command Organization is also composed of a Hanford Fire Department Operations Section Chief (assigned by the Incident Commander), trained support staff (e.g., Health Physics staff), and (as required) Hanford Fire Department medical responders and Hanford Patrol. In addition, personnel accountability aides are responsible for facilitating the implementation of protective actions (evacuation or take cover) and for facilitating the accountability of personnel after the protective actions have been implemented. Staging area managers are responsible for coordinating/conducting activities at the staging area. Personnel accountability aides assist the staging area managers by ensuring that personnel and visitors are properly evacuated from designated staging areas to a safe location. The Incident Command Organization supports actions requested by the Incident Commander and the BED.
A15.4.1.2 U.S. Department of Energy Hanford Emergency Operations Center. As stated in Section 15.4.1.3 of the SNF Project FSAR, the DOE Hanford EOC is an emergency response facility, maintained by the DOE, Richland Operations Office/Office of River Protection, to convene personnel providing essential emergency response functions. The DOE Hanford EOC will be activated for Alert and higher emergencies at the CSB.

A15.4.2 Assessment Actions

Provisions of Section 15.4.2 of the SNF Project FSAR cover development and implementation of the hazards surveys and hazards assessment, consequence assessments, and monitoring activities. These provisions apply to the CSB.

The CSB hazards survey will identify the conditions contained in the comprehensive emergency management program. A CSB emergency planning hazards assessment will be developed for the CSB for hazards that have the potential to generate an Alert or higher emergency. The hazards assessment will be prepared from the hazards survey and safety analyses that are developed and summarized in Chapter A3.0. The hazards assessment will also be derived from other pertinent facility documentation (e.g., safety assessment documents, interim safety basis documents, and special nuclear material accountability documents). The hazards assessment will provide the technical basis for the emergency management program. The scope and extent of planning and preparedness will directly correspond to the type and scope of hazards present and the potential consequences of events.

The hazards assessment will characterize the potential consequences on workers, the public, and the environment for each postulated accident and determine the emergency planning zone (EPZ) for each facility. The assessment also will determine the emergency class, protective actions, and the observable indications and criteria (emergency action levels) corresponding to the range of identified accidents.

A spectrum of potential accidents ranging from minor to beyond design basis are postulated and will be realistically analyzed for the CSB. While not every conceivable situation will be analyzed, the hazards assessments will provide the framework for response to virtually any declared emergency.

The methodology, assumptions, models, and evaluation techniques used in the hazards assessments are documented in Sections 15.4.2.1 and 15.4.2.2 of the SNF Project FSAR. Results from the CSB hazards assessment will be utilized in development of the building emergency plan for the CSB. Hazards assessments will be reviewed at least annually and updated, as necessary, in accordance with Section 15.4.2 of the SNF Project FSAR to delineate significant changes to the CSB or hazardous inventories and will be maintained in accordance with the Site contractor document control requirements.
A15.4.3 Event Categorization and Classification

Event categorization and classification for CSB emergency events will be accomplished in accordance with Section 15.4.3 of the SNF Project FSAR.

A15.4.4 Notifications

Notifications, in the event of an emergency event at the CSB, would be made in accordance with the provisions of Section 15.4.4 of the SNF Project FSAR in order to mitigate consequences and to protect the health and safety of workers, the public, and the environment.

A15.4.5 Emergency Facilities and Equipment

The building emergency plan for the CSB will be prepared and issued in accordance with HNF-IP-0263-GEN, Building Emergency Plan Guidance. A description of the facilities that will be available for coordinating CSB emergency response activities will be specified in the building emergency plan.

Emergency equipment consisting of materials and tools that may be required to measure, control, or mitigate the consequences of an emergency at the CSB is listed in Table A15-1. Detection ranges and types of instruments for radiological and nonradiological hazardous materials will be adequate for CSB emergency conditions as determined in Section A15.4.2. The emergency planning organization will ensure that sufficient emergency equipment is available. Location of this emergency equipment will be stated in the building emergency plan.

A15.4.6 Protective Actions

Protective actions are those actions that will be taken to preclude or reduce the exposure of individuals or the environment impacted by hazards or unsafe conditions during an emergency event at the CSB. These are presented in Section 15.4.6 of the SNF Project FSAR and are applicable to the CSB. Protective actions for the CSB will reflect the use of emergency response planning guidelines identified in Section 15.4.3.1 of the SNF Project FSAR. The planning guidelines published in the Emergency Response Planning Guidelines (AIHA 1988) will be used during a CSB emergency response to determine protective actions for unique exposures to chemical releases (see the SNF Project FSAR, Table 15-5). The protective action guides also are used during an emergency response to determine protective actions for unique exposures to radiological releases (see SNF Project FSAR, Table 15-2). DOE/RL-94-02, Hanford Emergency Management Plan, directs the use of the published protective action guides adopted by the states of Washington and Oregon.
<table>
<thead>
<tr>
<th>Type</th>
<th>Capabilities</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fixed and portable equipment</td>
<td></td>
</tr>
<tr>
<td>Fire control system</td>
<td>Assists in the control of a fire</td>
</tr>
<tr>
<td>- Fire detection and alarm system and wet</td>
<td>Assists in notifying personnel, summoning the Hanford Fire Department, and in fire suppression</td>
</tr>
<tr>
<td>pipe automatic sprinkler suppression system</td>
<td></td>
</tr>
<tr>
<td>Safety shower and eyewash station</td>
<td>Assists with personnel decontamination of hazardous (chemical) materials</td>
</tr>
<tr>
<td>Pressure alarms and/or water flow alarm</td>
<td>Assists with notifying personnel of emergency conditions</td>
</tr>
<tr>
<td>Evacuation and take cover siren</td>
<td>Assists with notifying personnel of emergency conditions and, by the type of siren, expected actions</td>
</tr>
<tr>
<td>Red crash alarm telephone</td>
<td>Alerts personnel in an emergency and communicates emergency information</td>
</tr>
<tr>
<td>Respiratory protection*</td>
<td>Protects personnel from hazardous chemicals</td>
</tr>
<tr>
<td>Portable emergency equipment</td>
<td></td>
</tr>
<tr>
<td>Fire extinguisher (Types A, B, and C)</td>
<td>Assists in fire suppression</td>
</tr>
<tr>
<td>Portable air compressor (brought in as needed)</td>
<td>Provides backup compressed air to failed compressed air system</td>
</tr>
<tr>
<td>Portable electrical generator (brought in as needed)</td>
<td>Provides backup electrical power where necessary to a facility that has lost normal electrical power</td>
</tr>
<tr>
<td>Hazardous materials spill control kits (unmounted)</td>
<td>Assists with hazardous (chemical) materials stabilization and cleanup following a spill or release</td>
</tr>
<tr>
<td>Command post equipment: emergency procedures, checklists (maps and</td>
<td>Provides area and site-specific emergency information</td>
</tr>
<tr>
<td>photographs of facilities optional)</td>
<td></td>
</tr>
<tr>
<td>Operational event scene equipment: radiological response vehicle,</td>
<td>Assists in controlling and mitigating the event</td>
</tr>
<tr>
<td>emergency procedures, duty cards, checklists, maps, photographs of</td>
<td></td>
</tr>
<tr>
<td>facilities</td>
<td></td>
</tr>
</tbody>
</table>
Table A15-1. Canister Storage Building Emergency Equipment. (2 sheets)

<table>
<thead>
<tr>
<th>Type</th>
<th>Capabilities</th>
</tr>
</thead>
<tbody>
<tr>
<td>Protective clothing and equipment</td>
<td></td>
</tr>
<tr>
<td>Anti-C clothing and personal protective</td>
<td>Provides contamination control (anti-C clothing for radiological and acid gear for any corrosive chemicals)</td>
</tr>
<tr>
<td>equipment</td>
<td></td>
</tr>
<tr>
<td>Miscellaneous respiratory equipment</td>
<td>Provides respiratory protection; this type of respirator equipment is not considered to be emergency equipment</td>
</tr>
</tbody>
</table>

*Respirators for emergency use will be thoroughly inspected at least once a month and after each use. Records of inspection dates and findings will be maintained.

The Hanford Site emergency management program uses the EPZ concept to focus emergency planning activities. EPZs are designated areas where protective actions may be required. The size of a zone is determined primarily by the expected dispersion distance of a particular concentration of a substance. The two exposure pathways for both radiological and nonradiological hazardous materials are the plume exposure pathway and the ingestion exposure pathway. See Section 15.4.6 of the SNF Project FSAR for a description of the exposure pathways.

The plume exposure pathway EPZ is the probable area of exposure to a passing cloud, or plume, of the substance potentially resulting in direct contact with the substance through the exterior of the body or through inhalation. The plume exposure pathway EPZ includes the area where emergency planning is conducted (1) to ensure that prompt and effective actions are taken in the event of an emergency, (2) to protect onsite personnel, and (3) to ensure public health and safety. The plume exposure pathway for the CSB (10 mi) is shown in Table 15-6 and Figure 15-5 in the SNF Project FSAR.

The ingestion exposure pathway EPZ is the probable area of exposure to contaminated foodstuffs or water potentially resulting in deposition of the material in various internal organs following ingestion (eating or drinking). The ingestion exposure pathway EPZ for radiological and nonradiological incidents at all Hanford Site facilities corresponds to the 80-km (50-mi) EPZ for Energy Northwest’s Nuclear Plant 2. The gray area in Figure 15-6 in the SNF Project FSAR represents the ingestion EPZ (80 km [50 mi]) for the Hanford Site.

The protective actions required to minimize the exposure of workers and the public are summarized in Section 15.4.6 of the SNF Project FSAR. Examples of protective actions as a function of accident category and consequences are illustrated in Table 15-7 in the SNF Project FSAR.
A15.4.7 Training and Exercises

An emergency organization for the CSB will be formed, trained, and tested in accordance with the provisions of Section 15.4.7 of the SNF Project FSAR. Drills and exercises will be developed in accordance with the provisions of Section 15.4.7 of the SNF Project FSAR with sufficient scope and detail to emphasize the facility-specific emergency events and response actions applicable to the CSB facility.

A15.4.8 Reentry and Recovery

The provisions applicable to a CSB emergency event termination, facility entry, transition from an emergency organization to a recovery organization, and the recovery process are provided in Section 15.4.8 of the SNF Project FSAR.

A15.5 DOCUMENT CONTROL

The building emergency plan, implementing procedures, reports of drills and exercises, and emergency event documentation for the CSB will be controlled and updated in accordance with the provisions of Section 15.5 of the SNF Project FSAR.

A15.6 REFERENCES


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CHAPTER A16.0

PROVISIONS FOR DECONTAMINATION AND DECOMMISSIONING
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   A16.3 DESCRIPTION OF CONCEPTUAL PLANS .......................... A16-1
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CSB       Canister Storage Building
D&D       decontamination and decommissioning
FSAR      final safety analysis report
HEPA      high-efficiency particulate air (filter)
HVAC      heating, ventilation, and air conditioning
MCO       multi-canister overpack
MHM       multi-canister overpack handling machine
SNF       spent nuclear fuel
A16.0 PROVISIONS FOR DECONTAMINATION AND DECOMMISSIONING

A16.1 INTRODUCTION

The provisions that apply to future decontamination and decommissioning (D&D) of Spent Nuclear Fuel (SNF) Project facilities are addressed in Chapter 16.0 of the SNF Project Final Safety Analysis Report (FSAR). Provisions specific to the Canister Storage Building (CSB) are addressed in this Chapter A16.0.

A16.2 REQUIREMENTS

The requirements that form the basis for D&D provisions are identified in Section 16.2 of the SNF Project FSAR.

A16.3 DESCRIPTION OF CONCEPTUAL PLANS

This section describes the design features and operational considerations pertinent to the CSB that will facilitate decontamination and ultimate decommissioning and environmental restoration activities. Design and operating practices that minimize the production of contamination and confine it to designated areas are important for final D&D considerations. General considerations applicable to all SNF Project facilities are provided in Section 16.3 of the SNF Project FSAR.

The multi-canister overpacks (MCOs), CSB, and MCO handling machine (MHM) are designed to prevent the spread of radioactive contamination from inside the MCOs. Baseline operations for the CSB ensure that the spread of radiological contamination from the MCOs or the transportation cask to the interior surfaces of the MHM, storage tubes, or CSB operating area is minimal during normal operations.

The fuel loading operation at the K Basins is designed to minimize contamination of the MCO and the cask–MCO during fuel loading. At K Basins, installing the immersion pail lid on the cask–MCO and inflating the immersion pail seals prevents pool water from coming in contact with the cask exterior and the cask–MCO interior cavity. The space between the pail and the cask and the cask cavity is filled with clean deionized water during in-pool loading operations to further prevent contaminants in the pool water from contacting the outside of the cask or entering the cavity between the MCO and the cask. After completing K Basin underwater fuel loading operations, the immersion pail is brought to the surface where all surfaces of the MCO shield plug top and immersion pail lid are rinsed with clean deionized water, surveyed, and decontaminated as required before the cask lid is installed.

At the Cold Vacuum Drying Facility, operators use a lid with inflatable seals to wash the cask exterior with tempered water before transporting the cask–MCO to the CSB.
A16.3.1 Design Features

If radiological contamination occurs, the CSB design features preclude the spread of contamination during normal operations and accident conditions. The four MCO access ports are sealed at the Cold Vacuum Drying Facility using gasketed, bolted cover plates before an MCO is received at the CSB. The initially selected MCOs will be monitored using the sampling/weld station equipment to ensure that pressure buildup in the MCO does not cause a hazardous condition and that the MCO is safe for interim storage at the CSB. The sampling/weld station hood limits the spread of contamination. After monitoring, a welded cap is installed on each MCO, providing additional protection to minimize the potential for release of radioactive contamination. In the overpressurization of an MCO (prior to having the cap welded on), the pressure is relieved using the sampling cart at the MCO sampling/weld station. The storage tube is designed to withstand all postulated design basis accidents and applicable natural phenomena hazard events listed in Chapter 1.0 of the SNF Project FSAR. The storage tube assembly is designed not to release contamination past the tube plug to the operating deck or into the vault. The operating deck area is isolated from the vault to prevent the migration of contamination from one zone to the other in the unlikely event of a release from an MCO. Deck area ventilation air is exhausted through a high-efficiency particulate air (HEPA) filter to the environment. All of the equipment, utilities, and operating stations that do not have a function in the operating area are isolated in the support area of the building to minimize risk of exposure to contamination. The support area is on a separate heating, ventilation, and air conditioning (HVAC) system from the operating area to minimize cross-contamination of zones. The support area is separated from the operating area by a shield wall with penetrations thoroughly sealed for maximum isolation.

Cracks and crevices have been minimized in the operating area to facilitate decontamination. Equipment used routinely to handle MCOs has seal welds to facilitate cleanup. Welded piping has been used where practical.

The design features that are of importance to D&D will be controlled through the design change control process to ensure that changes to the facility consider the impact on D&D.

A16.3.1.1 Load-In/Load-Out Area. MCOs are retrieved from or placed in transportation casks in the load-in/load-out area, located at the north end of the operating area. The equipment in the load-in/load-out area maintains secondary confinement as needed by operations to contain contamination in the unlikely event of a release from an MCO. The cask is placed in an open pit for MHM accessibility. During stages of the MCO handling process in which the cask is open and the MCO exposed, the area around the pit may be enclosed with a mobile service station enclosure. The concrete deck of the service station is covered with a coating that facilitates decontamination.

In the MCO service station with the mobile service station enclosure in place, the operating area atmosphere that could come into contact with the cask interior or MCO exterior is confined. The mobile service station enclosure is equipped with a HEPA-filtered exhaust system that maintains a negative pressure relative to the operating area and minimizes the potential for an
uncontrolled release of contamination to the operating deck. Thus, all air that exits the enclosure
is HEPA filtered before release to the operating area. The ventilation unit discharge is directed to
a nearby exhaust system collection point. The atmosphere inside the MCO (an abnormal MCO
may require servicing and purging) is HEPA filtered before being released to the operating area,
through the operating area ventilation system, and to the environment.

A16.3.1.2 Multi-Canister Overpack Handling Machine. The MHM provides a ventilated
chamber for transport of the MCO. In the MHM, the air atmosphere surrounding the MCO can
be maintained at a negative pressure relative to the operating area. Discharged exhaust is HEPA
filtered before release to the operating area. Any surface contamination released from the MCO
(as airborne contamination) will be filtered before it is released. The MCO provides the primary
confinement barrier for the SNF. The MHM serves as a secondary confinement when it contains
an MCO. The MHM has been designed to couple with equipment with which it performs an
MCO transfer function.

A16.3.1.3 Storage Tubes. The standard storage tubes have been designed as sealed vessels to
prevent particulate contamination from escaping. The overpack storage tubes have been designed
as sealed vessels to prevent contamination from escaping and to maintain an inert atmosphere.
The tubes are all welded construction. Welds having exposure to the MCO are seal welded,
including the bellows welds and the tube flange to the embed weld. With the tube plug in place,
the tube provides confined storage for the MCO. The standard tube plug has an O-ring seal, and
the overpack storage tube plug has double O-ring seals for the mating surface on the tube. The
standard tube plug has a filtered vent to minimize the potential for a release of contaminated
particulates. The overpack storage tube plug has a sealed port and a valved port. The overpack
storage tube is designed to contain releases from a stored MCO. A hold-down device for the
overpack storage tube plug prevents loss of gases if the overpack storage tube pressure increases.
During normal operations when a plug is removed from the storage tube, it is always contained in
a dedicated chamber within the MHM and has no opportunity to spread contamination.

A16.3.1.4 Sampling/Weld Area. The sampling/weld area comprises a row of seven process pits
and the equipment required for MCO canister cover assembly welding operations and monitoring
operations (sampling and monitoring). Each pit is recessed into the deck and covered. Welding
equipment is available for use at pits 2 and 7, and pits 3 and 6 provide helium gas and electrical
services to pits 2 and 7. Sampling and monitoring equipment is available for use at pit 7 for
monitoring and sampling the MCOs used in the monitoring process. Hoods are used during
sampling, monitoring, and welding operations to reduce the potential for release of contamination.
The pit cover has shield halves and a fitted center shield plate that allows working access to the
pit and serves a purpose similar to that of the tube plug. The design of the center shield plate and
shield halves enable the MHM to dock with the sampling/weld station, remove the center shield
plate, and perform MCO transfers while providing shielding for the operators. The portable,
ventilated, HEPA-filtered hoods (sampling and welding) maintain a negative pressure relative to
the operating area for contamination control. The hoods discharge to an area duct, which
discharges to the operating area ventilation system. The low volume of potentially contaminated
gases from an MCO being sampled or the minuscule amount of airborne contamination from the
welding process is further diluted by the approximately 100 ft³/min airflow through the hood into the exhaust duct. Filtration by the HEPA filter ensures that the length of ductwork exposed to potentially contaminated sample gases or weld fumes is kept to a minimum.

The center shield plate and shield halves must be removed, exposing the MCO, to perform sampling, monitoring, and welding operations. The portable hoods maintain a confined environment around the sampling, monitoring, and welding operations.

As discussed previously, the K Basin fuel loading operations use an immersion pail with inflatable lid seals, rinse with deionized water, survey the cask exterior, and decontaminate as required to ensure that contamination of the MCO exterior is as low as reasonably achievable. Radioactive airborne contamination, if any, should be minuscule when a canister cover assembly is welded to the MCO.

A16.3.1.5 Maintenance Pit. The maintenance pit is located on the east side of the load-in/load-out area. The pit will serve a number of uses relating to MHM and other equipment maintenance. The maintenance pit enables maintenance workers to gain access to all the serviceable components of the MHM. During recovery actions, it could serve as a work area for decontamination operations on the MHM and other equipment. Features associated with D&D activities include a sloped floor to a recessed pump-out pit and an industrial resinous coating on floor and wall surfaces.

A16.3.1.6 Electrical System. Electrical conduits and wiring cross between the operating area and the support area of the CSB. Sleeves have been included in the design for penetrations through the shielding wall. The sleeves facilitate the replacement of conduit should a contamination problem develop, and they also facilitate decommissioning. The pass-through openings of the sleeves are sealed to prevent the migration of contamination.

A16.3.1.7 Piping. Piping crosses between the operating area and the support area of the CSB. Sleeves have been included in the design for penetrations through the shielding wall. The sleeves facilitate the replacement of piping sections should a contamination problem develop and also facilitate decommissioning. The pass-through openings of the sleeves are sealed to prevent the migration of contamination. The piping systems in the operating area that have the potential for exposure to contamination include the inert gas, exhaust, and compressed air systems. The inert gas line is stainless steel and socket welded, which offers better resistance to external contamination. Piping in the pits is stainless steel for better decontamination. All piping in the service station area is socket welded.

A16.3.1.8 Tube Vent and Purge Cart. The tube vent and purge cart interfaces with the overpack storage tube plugs on an as-needed basis for abnormal or accident MCOs. In so doing, the cart breaks the sealed boundary of the tube and becomes the boundary for secondary confinement. The cart connects to the tube by means of a flexible metal hose. The contents of the storage tube are sampled, then if acceptable, may be vented to the operating area through a...
HEPA filter. All piping on the storage tube side of the HEPA filters is welded stainless steel. The cart piping will be pressure tested regularly to verify leak tightness.

**A16.3.1.9 Heating, Ventilation, and Air Conditioning System.** The HVAC system is designed to act, along with physical barriers, as a part of the CSB contamination confinement system to ensure contamination control within the facility. The HVAC system provides a controlled pressure gradient as air flows from outside the building inward through uncontaminated areas to potentially contaminated areas of the building, and out through HEPA filters and a monitored exhaust. The HVAC units are located in the HVAC equipment room of the support area building.

The operating area HVAC system is designed to maintain a negative pressure relative to the outside environment to ensure airflow from areas of lower contamination potential to areas of higher contamination potential. To prevent the possible migration of contamination to the outside of the building, ducts for outside air supply are equipped with back draft dampers to stop any backflow of air when the fans are not operating. Double doors provide an airlock at the north and south ends of the operating area to prevent loss of confinement during the movement of MCOs or equipment. The building air exchange rate is kept low (1.5 air changes per hour) by the use of local ventilation provided by the MHM, the mobile containment service tent, the tube vent and purge cart, and the sampling/weld area portable enclosures.

Each exhaust stream is HEPA filtered before release. Those streams discharged initially into the CSB operating area mix with the air in the operating area and are HEPA filtered a second time before discharge to the atmosphere. Exhaust duct work is welded stainless steel construction that minimizes contamination traps and provides an easily decontaminated surface. HEPA filters are located on short runs as close to the source as possible to minimize the amount of contaminated duct work. Bag-out filter housings are used to minimize contamination spread during filter changes except in the case of small specialty filters where bag-out filter housings are unavailable. Operating area ventilation system HEPA filters are changed out in the filter room, which has an airlock entrance that assists in containing the potential spread of contamination.

**A16.3.1.10 Personnel Contamination Control.** A room is provided in the support area building for surveying and decontaminating personnel. The facility design includes a dedicated change room for personnel working in radiation areas and potential contamination zones to keep contamination confined in a smaller area. An area within the support area building has been identified for step-off with access to the surveying/counting room and to the change room.

**A16.3.1.11 Concrete.** The likelihood of uncontained contamination being present in the CSB is small. The operating area floor is specified to have a trowel-finished class A finish. An industrial-grade chemical concrete sealer has been applied to its trowel-finished surface. The sampling/weld area also is coated with a chemical concrete sealer. The load-in/load-out floor area has an industrial resinous coating. This is sufficient to meet the intent of the requirement in ANSI/ANS-57.9-1992, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)*, paragraph 6.14(1), to provide concrete with an impervious surface. Metal liners for pits...
(maintenance pit and Fast Flux Test Facility pit) and the service station area were considered but are not included in the design. Metal liners and additional concrete coatings may be added to the facility if enhancement of decontamination capability is required for future operations.

A16.3.1.12 Safety Features. Chapters A3.0, A4.0, A5.0, and A6.0 identify the selected safety features for preventing or mitigating the postulated accidents (see tables identified in Chapter A3.0 for each accident). These implemented engineered barriers and administrative programs will prevent or greatly reduce the consequences of any postulated abnormal event or accident, thus the amount and location of any radiological material releases inside the CSB will be diminished for D&D cleanup activities.

A16.3.2 Operational Considerations

Because baseline operations assume no spread of contamination from the transportation cask or MCO to the operating area of the CSB, no special facilities for the support of decontamination activities have been provided. Although the risk of contamination is minimal, operating procedures address requirements for radiological control surveys during stages of the MCO handling process to determine whether an MCO is leaking contamination and to prevent its spread. The load-in/load-out area and mobile service station enclosure, as well as the top of the sampling/weld pits and the top of the storage tubes and plugs, may require survey and decontamination following use.

The potential for personnel, equipment, and building contamination within the CSB is minimized by the design of the facility and equipment and by administrative controls, radiological contamination surveys, and work guidelines defined in operating procedures and work permits. The general design features and operating practices described in Sections 16.3.1 and 16.3.2 of the SNF Project FSAR were considered in the design of the CSB to minimize the spread of contamination, simplify D&D operations, and help minimize site and environmental contamination.

During CSB operations, site environmental contamination is minimized because of the following protection features:

- No expected liquid releases
- HEPA-filtered final exhaust
- Sealed MCOs
- Negative building pressure maintained by HVAC system.

Following the initial vault loading phase, no activities are expected to occur that would contaminate CSB facility components or structures. The CSB facility will continue to receive periodic surveillance and maintenance as required to ensure that critical items listed above required to support D&D activities are operable.
A16.3.3 Decommissioning

The construction of the operating areas of the CSB superstructure as metal frame and siding, and the vault and deck as concrete, will simplify decontamination and dismantling. The CSB has no floor drains or drain system but operates with collection sumps. This eliminates the possibility of a contaminated drain system to contend with in decommissioning. The CSB and its process equipment may be removed at a future time in compliance with applicable regulations. This could be required when equipment is obsolete or there is no continuing need for the facility. A program would be established at that time to address the decontamination required for decommissioning of the CSB and its related process equipment to eliminate the need for active maintenance of that site.

Conceptual plans for D&D will include an updated facility hazard analysis for the D&D activities, which will be used to prepare the plan to administer the expected D&D strategy. Conceptual plans also will include a preliminary deactivation plan that contains at least the following information:

- Structures, systems, and components in their final configuration
- Review and determination of status for structures, systems, and components based on mission and life cycle phase
- Configuration management for missing or inaccurate design baseline documentation, voiding and downgrading of design documents, and turnover of design baseline documents to the environmental restoration contractor.

If the CSB is identified for future use, an active maintenance plan will be instituted in place of the decommissioning plan. Decommissioning plans for the CSB facility will be developed and reviewed against the existing environmental impact statement. The environmental impact statement will be updated to include D&D activities in accordance with the National Environmental Policy Act (NEPA) of 1969 process if deemed appropriate. See Section 16.3.3 of the SNF Project FSAR for the process used to develop the CSB D&D plan.

A16.4 REFERENCES


National Environmental Policy Act (NEPA) of 1969, 42 USC, 4321, et seq.
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CHAPTER A17.0

MANAGEMENT, ORGANIZATION, AND INSTITUTIONAL SAFETY PROVISIONS
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<tr>
<td>CSB</td>
<td>Canister Storage Building</td>
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<tr>
<td>FSAR</td>
<td>final safety analysis report</td>
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A17.0 MANAGEMENT, ORGANIZATION, AND INSTITUTIONAL SAFETY PROVISIONS

A17.1 INTRODUCTION

A description of the organizational structure, responsibilities, and interfaces that support safe design, construction, and operational activities of the Canister Storage Building (CSB) as a subproject of the Spent Nuclear Fuel (SNF) Project are addressed in Chapter 17.0 of the SNF Project Final Safety Analysis Report (FSAR).

A17.2 REQUIREMENTS

The requirements that form the basis for management, organizational, and safety provisions are identified in Section 17.2 of the SNF Project FSAR.

A17.3 ORGANIZATIONAL STRUCTURE, RESPONSIBILITIES, AND INTERFACES

The overall structure, responsibilities, and interfaces for CSB operations are identified in Section 17.3 of the SNF Project FSAR.

A17.4 SAFETY MANAGEMENT POLICIES AND PROGRAMS

The safety management policies and programs applicable to CSB are identified in Section 17.4 of the SNF Project FSAR.
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