March 5, 2001

The Honorable Carolyn L. Huntoon
Acting Assistant Secretary for
Environmental Management
Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585-0113

Dear Dr. Huntoon:

The Hanford Spent Nuclear Fuel Project (SNFP) is critical to resolving the spent fuel vulnerabilities identified in the Defense Nuclear Facilities Safety Board's (Board) Recommendation 94-1, *Improved Schedule for Remediation in the Defense Nuclear Facilities Complex*. The project provides for the removal, conditioning, and interim dry storage of the deteriorating spent N-Reactor fuel stored underwater in the aging basins at the K-Reactors.

The Board previously forwarded a technical report, DNFSB/TECH-17, *Review of the Hanford Spent Nuclear Fuel Project*, October 1997, addressing the schedule problems associated with the SNFP at Hanford. Since DNFSB/TECH-17 was issued, the Board's staff has continued its reviews of the project to ensure that safety problems are identified and addressed expeditiously and effectively. The results of these reviews are described in the enclosed technical report, DNFSB/TECH-30, *Safety Review of the Hanford Spent Nuclear Fuel Project During the Design and Construction Phase*, November 2000.

The Board believes that the valuable lessons learned on this project in areas such as quality assurance, preoperational testing, phased Safety Analysis Report preparation, and design reviews should be applied to the ongoing project efforts for the K-East Basin. They should also be of value for application to other projects throughout the defense nuclear complex.

This report is transmitted for your consideration and potential use on other projects under the cognizance of the Office of Environmental Management, as well as in continuing assessments of the SNFP at Hanford.

Sincerely,

John T. Conway
Chairman

Enclosure

c: Mr. Mark B. Whitaker, Jr.
Mr. Keith A. Klein

DNFSB/TECH-30

SAFETY REVIEW
OF THE
HANFORD SPENT NUCLEAR FUEL PROJECT
DURING THE
DESIGN AND CONSTRUCTION PHASE

Defense Nuclear Facilities Safety Board
Technical Report

February 2001

DNFSB/TECH-30

SAFETY REVIEW OF THE HANFORD SPENT NUCLEAR FUEL PROJECT DURING THE DESIGN AND CONSTRUCTION PHASE

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EXECUTIVE SUMMARY

The need for safe, expeditious stabilization of hazardous residual materials from the nation's nuclear weapons program was identified by the Defense Nuclear Facilities Safety Board (Board) in its Recommendation 94-1, *Improved Schedule for Remediation* (Defense Nuclear Facilities Safety Board, 1994). A key element of the Department of Energy's (DOE) Implementation Plan for Recommendation 94-1 (O'Leary, 1995), is the Spent Nuclear Fuel Project (SNFP) at the Hanford Site.

The Board's staff completed a review of schedule problems associated with the SNFP in the fall of 1997; the results of that review were documented in a technical report, DNFSB/TECH-17, *Review of the Hanford Spent Nuclear Fuel Project* (Defense Nuclear Facilities Safety Board, 1997). After DNFSB/TECH-17 was issued, the Board's staff continued its reviews of the project to ensure that safety problems were identified and addressed expeditiously and effectively.

Applying the key tenets of Integrated Safety Management, the Board's staff reviewed the SNFP activities through frequent site visits, technical meetings, and document reviews, as well as ongoing reviews by the Board's Site Representatives assigned full-time to the Hanford Site. Issues identified during these reviews, and DOE actions to resolve them include:

- Additional design features and procedural controls to prevent and mitigate potential runaway thermal reactions (rapid oxidation of the irradiated spent fuel leading to an uncontrolled thermal reaction and ignition of the fuel). (see Section 3.2.2.2)

- Additional actions to reduce the potential and mitigate the consequences of a cask drop event in the basin. (See Section 3.2.3.3)

- Installation of an annunciator to indicate a loss of battery room ventilation. (See Section 3.2.3.4)

- Enhanced quality assurance of the Multi-canister Overpack through appropriate application of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. (See Section 3.3.3)

- Installation of reliable standby power in the Cold Vacuum Drying Facility for safety-significant ventilation. (See Section 3.5.3.8)

- Installation of a lightning protection system for the ventilation exhaust stack at the Cold Vacuum Drying Facility. (See Section 3.5.3.8)

- Correction of design analysis deficiencies identified through a confirmatory structural analysis. (See Section 3.6.3.1)

The issues described above were resolved satisfactorily by DOE and its contractors working with the Board and its staff. The resolution of these issues at the K-West Basin provide valuable lessons to be applied to the K-East Basin and to other projects throughout the defense nuclear complex. These lessons include:

- *Phased Development of Safety Analysis Documentation* — A comprehensive Preliminary Safety Analysis Report is necessary to minimize changes to design, component classification, and safety assumptions. (See Section 4.1)

- *Design Reviews*—Thorough, timely design reviews are essential for ensuring early identification of inadequate design features. (See Section 5.2)

- *Quality Assurance*—Each phase of design, procurement, fabrication, and construction must rigorously apply to quality assurance standards and practices to avoid costly and time consuming deficiencies. (See Section 4.3)

- *Preoperational Testing*—Preoperation test programs must ensure the appropriate rigor in conducting and documenting tests, must include integrated tests rather than relying solely on tests of components and subsystems, and must include sufficient schedule to allow recovery from failures or deficiencies. (See Section 4.6.2)

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1. INTRODUCTION

1.1 BACKGROUND

The need for safe, expeditious stabilization of hazardous residual materials from the nation's nuclear weapons program was identified by the Defense Nuclear Facilities Safety Board (Board) in its Recommendation 94-1, *Improved Schedule for Remediation* (Defense Nuclear Facilities Safety Board, 1994). The importance of this matter was underscored by commitments contained in the Department of Energy's (DOE) Implementation Plan for Recommendation 94-1 (O'Leary, 1995), which was accepted by the Board.

The Spent Nuclear Fuel Project (SNFP), a key element of DOE's Implementation Plan for Recommendation 94-1, is one of the most critical projects at the Hanford Site. The project addresses the spent nuclear fuel currently stored in the Hanford K-Basins. During the Cold War, fuel discharged from the Hanford production reactors was processed in the Plutonium/Uranium Extraction Facility (PUREX) to recover the plutonium created by neutron irradiation. PUREX operated in this mode from 1956 until 1972. It was shut down from 1972 to 1983, then operated for about 7 years to process irradiated N-Reactor fuel. In 1990, PUREX was placed in standby status, and in 1992 it was shut down permanently, thus eliminating processing capabilities that could have been used to stabilize the substantial inventory of spent fuel in the K-Basins. This remaining spent fuel has been stored for an extended period and its condition has deteriorated, some of it seriously.

During September 1997, the Board's staff completed a review of serious schedule problems associated with SNFP that threatened timely completion of this important risk-reduction effort. The results of that review were discussed in a letter from the Board dated November 18, 1997 (Conway, 1997), which enclosed a technical report on the subject, DNFSB/TECH 17, *Review of the Hanford Spent Nuclear Fuel Project* (Defense Nuclear Facilities Safety Board, 1997).

Since DNFSB/TECH-17 was issued, the Board's staff has continued its reviews of the project to ensure that safety problems are identified and addressed expeditiously and effectively. Although the 1997 schedule slippage has not been recovered, the project has been aggressively managed to a new baseline established in 1998. Significant changes in project management have occurred, and key management personnel have been added to the contractor organization. These changes have addressed a key weakness identified in DNFSB/TECH-17 that challenged the safe and timely completion of the project. Since issuance of DNFSB/TECH-17, identification and evaluation of problems have been performed more systematically, and specific actions to address and resolve problems have been identified and institutionalized.

As part of its ongoing review of the project, the Board's staff has reviewed SNFP activities in the field through frequent site visits, as well as through ongoing monitoring of the project by the Board's Site Representatives assigned full-time to the Hanford Site. This report presents the results of the staff's reviews of the design and construction activities of the SNFP and the associated safety documentation, as well as extensive interaction with DOE personnel and the SNFP contractors, to provide assurance that the SNFP facilities can be operated safely.

1.2 ORGANIZATION OF THIS REPORT

The organization of the major portion of this report reflects the staff's application of the key tenets of Integrated Safety Management (ISM) to the design and construction phase of a facility's life cycle. That is, the reviews on which the report is based addressed each of the five core functions of a sound ISM System, as outlined in DNFSB/TECH-16, *Integrated Safety Management* (Defense Nuclear Facilities Safety Board, 1997). ISM represents a common-sense approach to planning and performing virtually all types of hazardous work, applying as well to the design and construction phases of the facility life cycle as to the operations phase. These core functions, adapted to the design and construction phases of the facility life cycle (see Figure 1-1), are as follows:

- Definition of the work and how it is to be accomplished (i.e., mission statement and identification of functional requirements),

- Analysis of the hazards entailed in performing the work,

- Identification and description of the controls (structures, systems, and components [SSCs]) and administrative controls necessary to accomplish the facility mission safely,

- Performance of design and construction work within structured programs for ensuring implementation of hazard controls, using adequately trained personnel, and

- Assessment of how well the work was performed, including provisions for feedback and continuous improvement.

Section 2 provides an overview of the SNFP mission, as well as a general description of the sequence of activities and the relationships among the facilities and processes involved in the project. This information sets the stage for the observations and findings resulting from the staff's reviews, which are presented in the later sections.

The above core functions serve as the framework for presentation of the results of the staff's reviews in Sections 3, 4, and 5.

Section 3 addresses each of the following SNFP facilities and major activities:

- Project approach to facility level hazard analysis and controls (Section 3.1),

- Systems and activities in the K-Basins (Section 3.2),

- Design and fabrication of the Multi-canister Overpacks (MCOs) (Section 3.3),
- Transport of the MCOs between facilities (Section 3.4),
- Design of systems and conduct of activities in the Cold Vacuum Drying Facility (Section 3.5), and
- Design of systems and conduct of activities in the Canister Storage Building (Section 3.6).

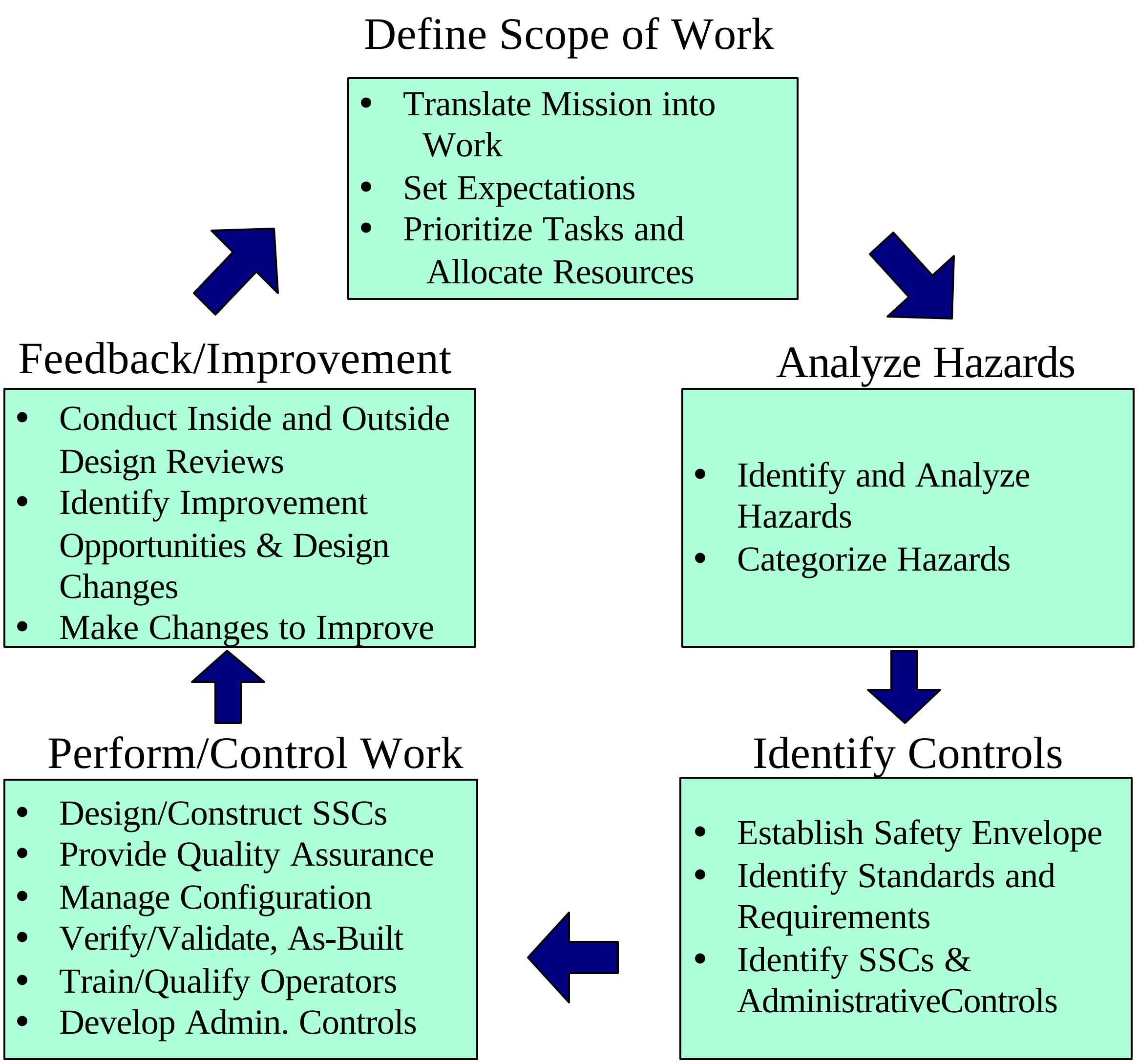

Figure 1-1. Safety Management Functions: Design and Construction Life-Cycle Phase

Source: DNFSB/TECH-16, *Integrated Safety Management* (Defense Nuclear Facilities Safety Board, 1997).

For each of the above facilities and activities covered in Section 3, the first three core ISM functions (definition of the scope of work, analysis of the hazards posed by the work, and identification and description of needed controls) are addressed, subsystem by subsystem.

Section 4 addresses the fourth core ISM function, associated with those programs that ensure that hazard controls will be properly implemented, including the process for establishing activity level controls for worker protection.

Section 5 describes project-level activities designed to ensure that the final core ISM function, feedback and improvement, has been adequately addressed.

Finally, Section 6 presents conclusions reached by the Board's staff, based on its reviews and evaluations of the SNFP design activities and supporting documentation.

A glossary of abbreviations and acronyms and a list of references are provided at the end of the report.

2. OVERVIEW OF THE SPENT NUCLEAR FUEL PROJECT

2.1 MISSION

The mission of the SNFP is to resolve the safety and environmental issues associated with continued wet storage of deteriorating spent nuclear fuel (SNF) in the Hanford K-Basins. Accomplishing this mission involves several major functions conducted primarily in three separate facilities:

- Preparing and loading fuel into Multi-canister Overpacks (MCOs) in the K-Basins,
- Transporting wet MCOs from the K-Basins to the Cold Vacuum Drying Facility (CVDF),
- Draining free water from the MCOs and drying them in the CVDF,
- Transporting loaded MCOs from the CVDF to the Canister Storage Building (CSB), and
- Storing loaded MCOs on an interim basis in the CSB facility, pending ultimate disposal in a suitable repository.

The K-Basins and the CVDF are located in the 100K Area of the Hanford Site (see Figure 2-1); the CSB is approximately 8 miles away in the 200 East Area and is more than 250 feet above the design basis flood level of the Columbia River (see inset in Figure 2-1).

2.2 PROCESS AND FACILITY OVERVIEW

The functions and processes associated with each major SNFP facility are briefly described below. Portions of the descriptions of facilities and the figures in this section are taken from the safety basis documents for the facilities.

2.2.1 K-Basins

The K-West (K-W) and K-East (K-E) Basins are large pools of water originally used to store and handle SNF discharged from the K-Reactors as part of the weapons material production effort. During the late 1970s and early 1980s, the K-Basins were modified to store SNF from the N-Reactor as well.

The basin water provides radiation shielding and removal of heat generated by decay of the fission products in the SNF. The water is continually filtered and treated to improve visibility, and is cooled to remove the heat generated by radioactive decay of the SNF and to reduce the rate of corrosion of the fuel elements. The basins currently contain about 2,270 metric tons of SNF, consisting of approximately 105,000 N-Reactor fuel elements.

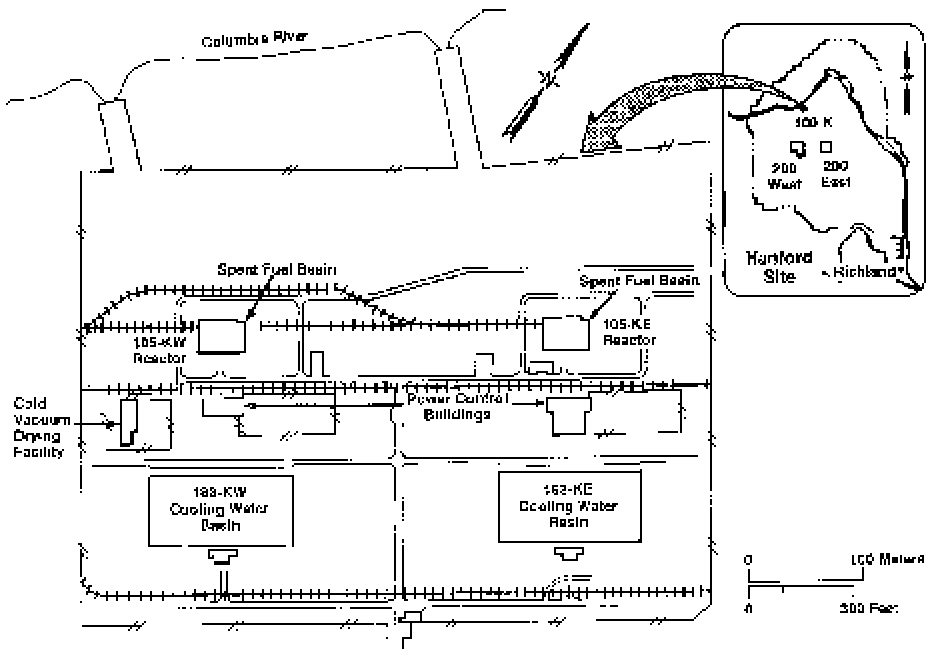

Figure 2-1. 100 K Area

Source: Fluor Hanford, Inc., HNF-WM-SAR-62, Revision 4 (Fluor Hanford, Inc., 2000).

Principal activities to be conducted in the K-Basins in preparation for removal of the SNF involve fuel retrieval; cleaning and sorting of fuel assemblies; loading of fuel and sludge into baskets, then into MCOs; and water treatment to remove accumulated sludge and soluble contaminants. The following modifications have been made to the K-W Basin to permit safe packaging and removal of SNF. (The systems described below were installed in the K-W Basin; similar changes are planned for later installation in the K-E Basin.)

- *Fuel Retrieval System* (FRS)—The FRS provides the capability to clean, sort, and load spent fuel elements into the MCO baskets. The primary functions of the FRS are to retrieve the SNF assemblies, separate them from the canisters where they presently reside, clean the assemblies to meet downstream processing requirements, load the SNF into MCO baskets, and position the MCO baskets for loading into MCOs. Remotely operated manipulators are used for sorting and loading the SNF to reduce worker exposure to radiation or contamination.

- *Integrated Water Treatment System* (IWTS)—The IWTS maintains basin water quality during fuel removal activities and provides water directly to the basin and fuel removal processes. The IWTS consists of submersible pumps drawing suction from three locations in the fuel retrieval system, then routing the water through a series of particulate and sludge removal vessels and an ion exchange module to remove soluble radionuclides (primarily cesium) before returning the water to the basin or, as required by operational needs, distributing it to other basin users (e.g., MCO South Loadout Pit flush, MCO transfer cask rinse). The IWTS is managed as a closed-loop system and can be coupled with the existing skimmer system to provide increased operational flexibility.

! *Cask Loadout System* (CLS)—The CLS includes the MCO Loading System, used to place the loaded baskets of SNF into the MCO, which is inside a transfer cask in the South Loadout Pit. An immersion pail protects the outside of the transfer cask from contamination when it is immersed in the basin. The Immersion Pail Support Structure is the primary load-carrying structure and interfaces directly with the South Loadout Pit. The Operator Interface Platform provides operator access near the center of the South Loadout Pit during SNF loadout operations.

The basin and supporting areas are located directly north of the 105 K-Reactor building (see Figure 2-2). The main basin includes three bays separated by vertical concrete walls. Several pits—including the Dummy Elevator Pit, Weasel Pit, Technical View Pit, and North and South Loadout Pits—and the reactor fuel discharge chute are attached to the main basin. There is a center island between the main basin and the discharge chute. Isolation barriers installed between the basin walls and the center island separate the basin from the discharge chute to mitigate the effects of a failure of the construction joint between the discharge chute and the reactor building. A transfer bay with a large crane is located west of each basin to support fuel-handling activities.

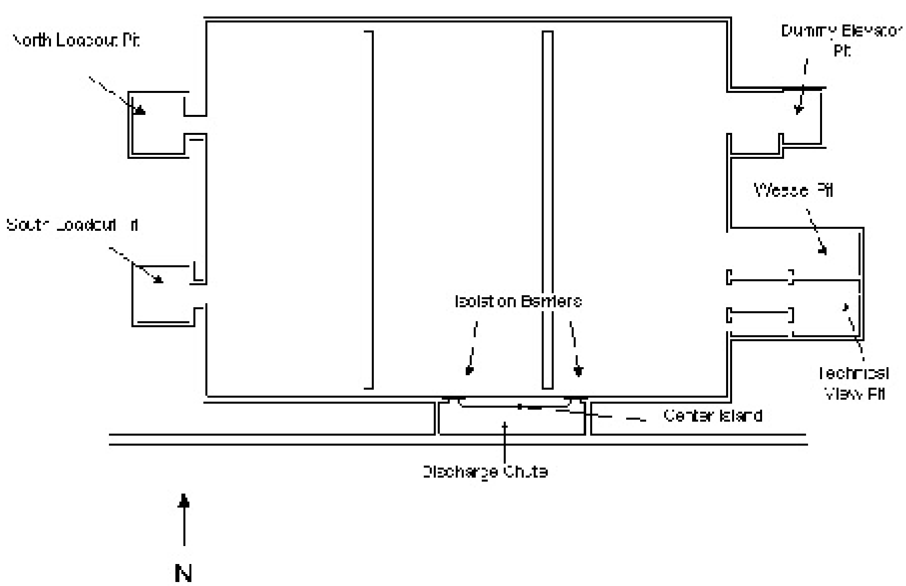

Figure 2-2. Plan View of K-West Basin

Source: Fluor Hanford, Inc., HNF-WM-SAR-62, Revision 4 (Fluor Hanford, Inc., 2000).

Essentially all of the spent fuel currently stored in the K-Basins is from the N-Reactor. A small amount of single-pass reactor SNF is stored in the basins; this SNF is not included in the Final Safety Analysis Report. An addendum for the SAR is planned before this fuel is moved from the basin. A typical N-Reactor Mark IV fuel element is shown in Figure 2-3. Ongoing operations now under way involve surveillance, monitoring, and water treatment as required to maintain proper storage conditions, and control and accountability of special nuclear materials (safeguards). These activities are covered by an existing authorization basis agreement, based on appropriate hazard and safety analyses.

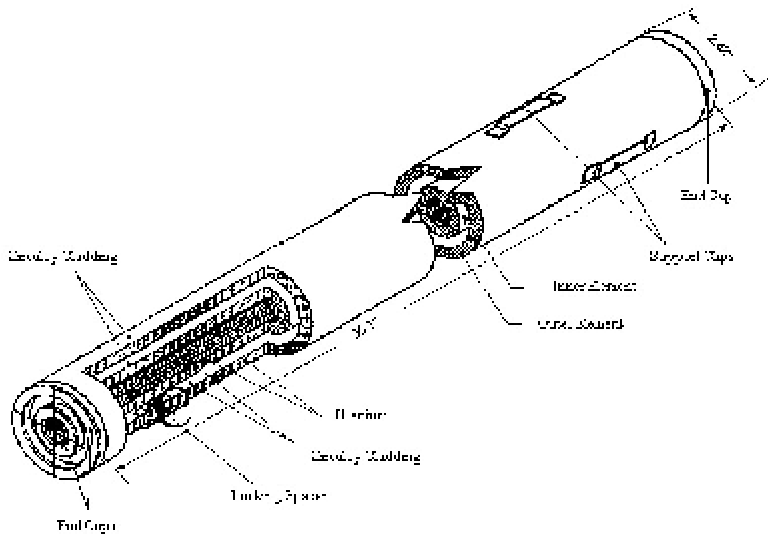

Figure 2-3. N-Reactor Fuel Assembly (Mark IV)

Source: Fluor Hanford, Inc., HNF-SD-TP-SARP-017, Revision 2 (Fluor Hanford, Inc., 2000).

2.2.2 Multi-canister Overpack

The safety function of the MCO is to provide containment of the SNF during stabilization and storage operations (see Figure 2-4). The MCO is a stainless steel canister approximately 24 inches in diameter and almost 14 feet long (with its cover assembly installed), constructed to Section III, Division 1, Subsection NB, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 1998). A 150 pounds per square inch, gauge (psi[g]) rupture disc in the shield plug assembly limits internal MCO pressure during processing. After processing, welding of the cover assembly to the MCO, which is done in the CSB, encloses the rupture disk and access ports, providing a totally welded, sealed container with a design pressure of 450 psi(g). Pressurization post-closure is not anticipated.

As part of the activities conducted in the K-Basins, cleaned SNF is placed in MCO fuel or scrap baskets. The design of fuel and scrap baskets differs, depending on the fuel enrichment (Mark IA fuel at 1.25 percent; Mark IV at 0.95 percent) and fuel element length. Five or six fuel baskets, depending on the type of N-Reactor fuel elements involved, are then stacked inside an MCO. Five baskets per MCO are used for Mark IV fuel elements; six baskets are used for the shorter Mark IA elements. Scrap baskets are used to hold pieces of fuel elements that cannot be stacked in a fuel basket. When they are used, scrap baskets replace fuel baskets in the MCO.

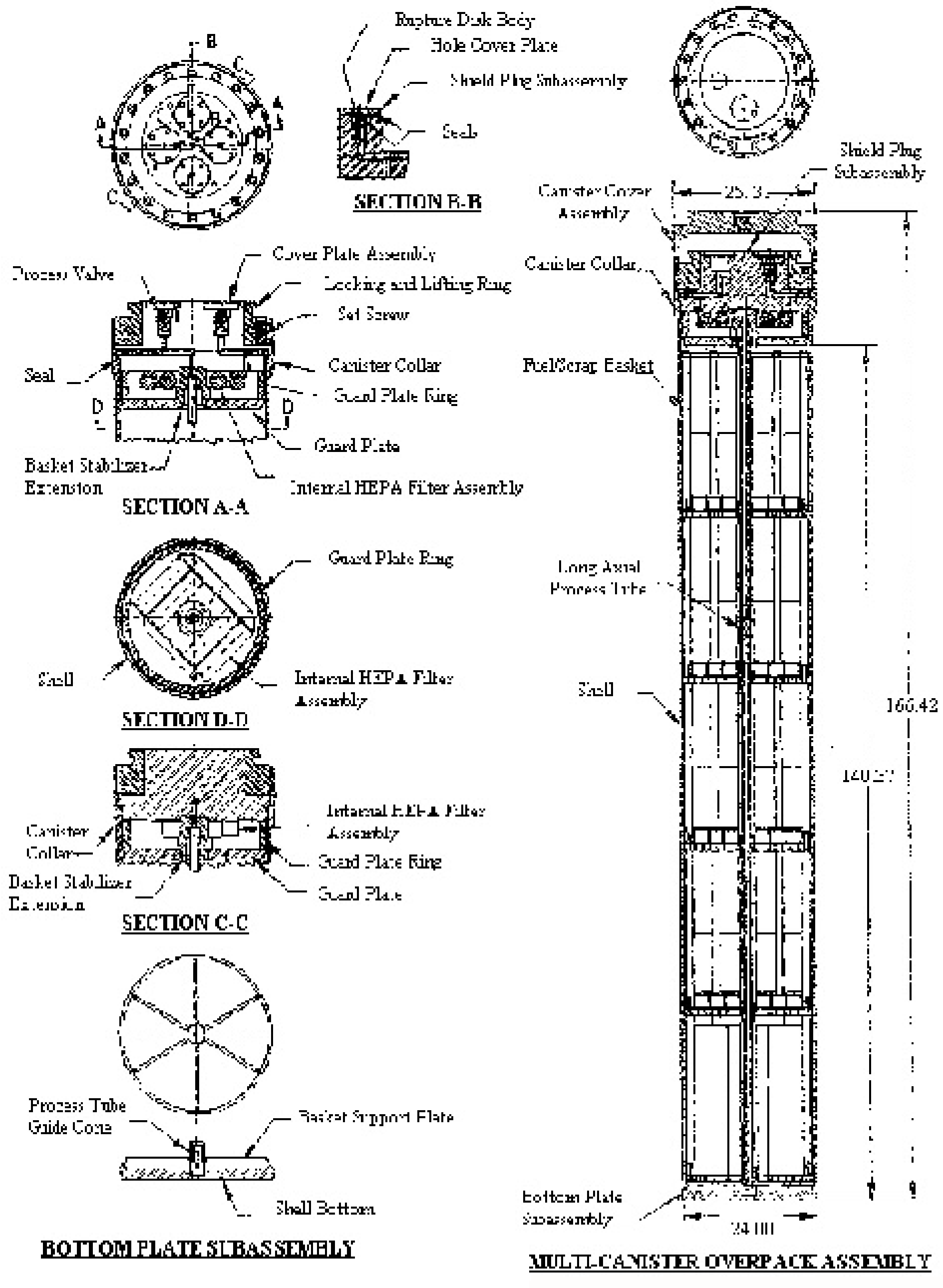

Figure 2-4. Multi-canister Overpack Assembly

Source: Fluor Hanford, Inc., HNF-3553, Annex A, Revision 0 (Fluor Hanford, Inc., 2000).

During loading of the fuel and scrap baskets, the MCO is inside a shielded transfer cask (the MCO transfer cask also serves as a transportation cask) in the K-Basin South Loadout Pit. After the baskets are loaded into the MCO, a shield plug assembly is installed in the MCO, using a mechanical seal between the shield plug assembly and the MCO shell.

After the transfer cask cover is installed, the loaded MCO transfer cask is removed from the South Loadout Pit and positioned on a truck trailer. The loaded MCO is then transported to the CVDF, where the MCO and its contents are dried (see Section 3.5). After drying in the CVDF, the MCO is transported to the CSB facility. There it is placed in a storage tube located in a below-grade structure for interim storage, pending ultimate disposal in a repository (see Section 3.6).

2.2.3 MCO Transfer Cask and Transporter

The safety functions of the Cask Transportation System (see Figure 2-5) are to provide shielding from radiation emitted by the highly radioactive fuel elements inside a loaded MCO, and to provide protection and containment of the MCO during on-site movements, including possible accidents. The system consists of a shielded MCO transfer cask and a tractor-trailer transporter capable of moving MCOs safely between facilities, and of serving as a temporary storage and handling device for MCOs during dewatering activities conducted at the CVDF.

The MCO transfer cask is a vertical stainless steel cylinder approximately 40 inches in diameter and 170 inches high with a bolted and sealed stainless steel lid and integral lifting device providing containment; the cylinder walls are 7.31 inches thick, providing radiation shielding. The MCO transfer cask is transported on a trailer designed specifically for this purpose and used only for on-site transportation. For processing in the CVDF, a work platform is provided on the trailer for access to the top of the MCO with the cask lid removed.

2.2.4 Cold Vacuum Drying Facility

The safety functions of the CVDF are to remove basin water from the MCO while maintaining the integrity of the containment boundary and preventing runaway thermal reactions due to rapid oxidation of the fuel. Bulk water is removed from the MCO using pressurized helium to force the water out through an installed dip tube. Additional free water is removed through cold vacuum drying performed at less than 0.1 pounds per square inch, absolute, and at 50° centigrade for a minimum processing time of 50 hours. After cold vacuum drying, no more than 0.5 pounds of free water is expected to remain in the MCO.

The CVDF, shown in Figure 2-6, is a newly constructed facility, approximately 230 feet long, 80 feet wide, and 35 feet high. The process bay area (60 feet wide by 150 feet long by 35 feet high) contains five bays. Bay 1 is used as a spare bay without services or equipment. Only two of the bays (Bays 4 and 5) are fully functional as process bays. Bay 3 has a process equipment skid in place, but the skid is not connected and will be used for equipment spares. Bay 2 contains a second-level mezzanine but no process equipment. The initial CVDF design used four process bays to meet production schedules; however, experience with operations modeling and testing indicated that only two process bays would be needed. The design of the process bay area consists of a steel frame with attached concrete panels to facilitate decontamination and demolition.

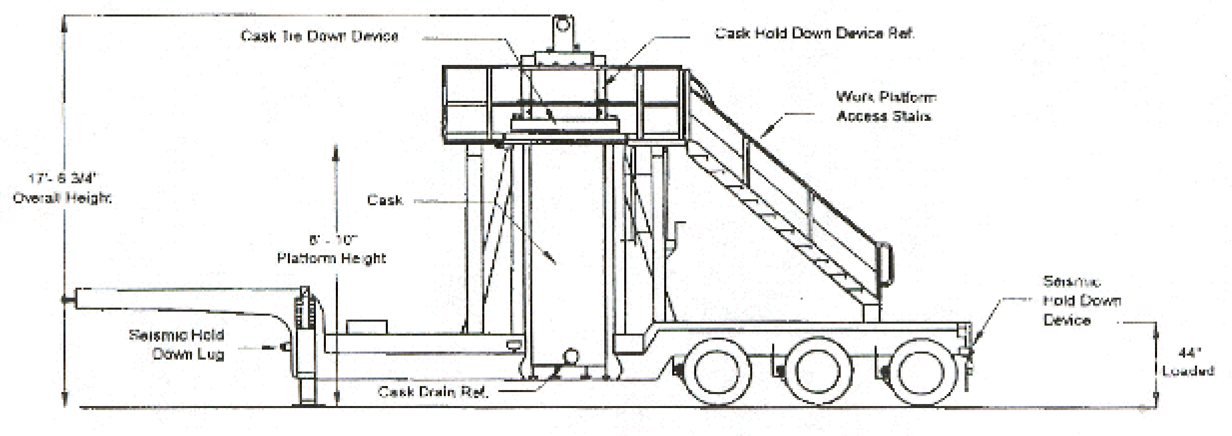

Figure 2-5. Cask Transportation System

Source: Fluor Hanford, Inc., HNF-SD-TD-SARP-017, Rev. 2 (Fluor Hanford, Inc., 2000)

The process support area (20 feet by 150 feet) 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 adjoins the north wall of process Bay 1. It is constructed as a single-story, steel frame building with 10-inch-thick exterior walls of precast concrete panels. The administration building, adjacent to process Bay 5, is a single-story, preengineered metal building with an exterior wall of insulated metal panels. All roofs are metal. The CVDF exhaust stack (48 feet high and 30 inches in diameter) is located 17 feet from the west wall of the process support area.

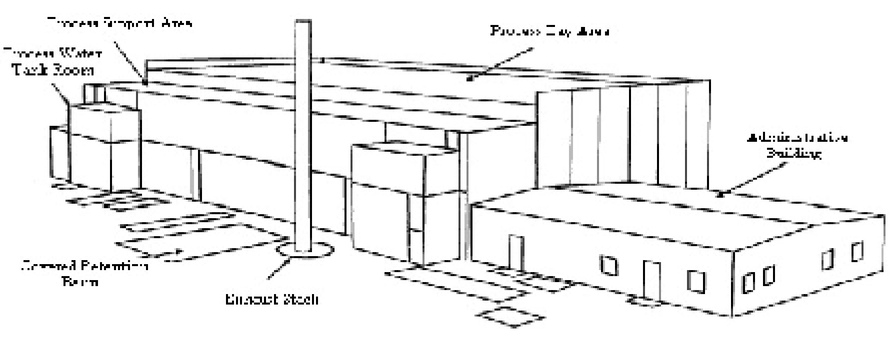

Figure 2-6. Isometric View of Cold Vacuum Drying Facility (diesel generator building not shown)

Source: Fluor Hanford, Inc., HNF-3553 Annex B, Revision 1 (Fluor Hanford, Inc., 2000).

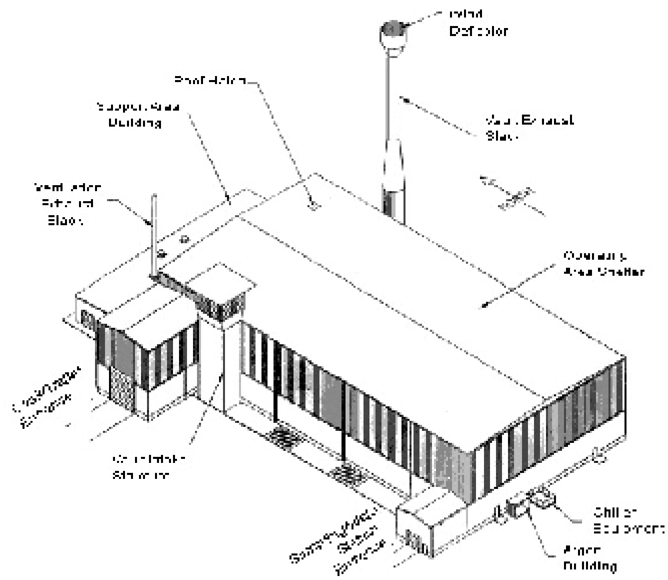

Figure 2-7. Isometric View of Canister Storage Building

Source: Fluor Hanford, Inc., HNF-3553, Annex A, Revision 0 (Fluor Hanford, Inc., 2000).

2.2.5 Canister Storage Building

The safety functions of the CSB are to receive and transfer MCOs within the building, store them in below-grade storage tubes; provide a shielded sampling/weld station; and maintain adequate shielding, passive cooling, and containment of the MCOs during interim storage.

The CSB is a newly constructed facility, consisting of a steel frame building enclosing the operating area, the loading/loadout area, and three below-grade concrete vaults of equal size (see Figures 2-7 through 2-9). The vaults are covered by a concrete operating floor. Support functions and equipment are housed in a steel frame 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 storage tubes for staging of mechanically sealed MCOs and for interim storage of the MCOs with welded cover assemblies.

The operating floor of the CSB is a reinforced concrete structure approximately 230 feet long (north-south) by 137 feet wide (east-west). The operating floor is bounded to the north by the loading/loadout area (trailer vestibule and MCO service station) and support area building foundations, and to the south by the sampling/weld area foundation. The operating floor structure contains numerous full-thickness embedded steel sleeves that receive the 220 storage tubes and tube plugs for the standard storage tubes and 6 overpack storage tubes in Vault 1. Vault 1 is air-cooled, using natural convection. Air enters the cooling system through an above-grade inlet structure, then flows through a below-grade concrete intake plenum, passing around the outside of the storage tubes to an exhaust plenum and out the exhaust stack.

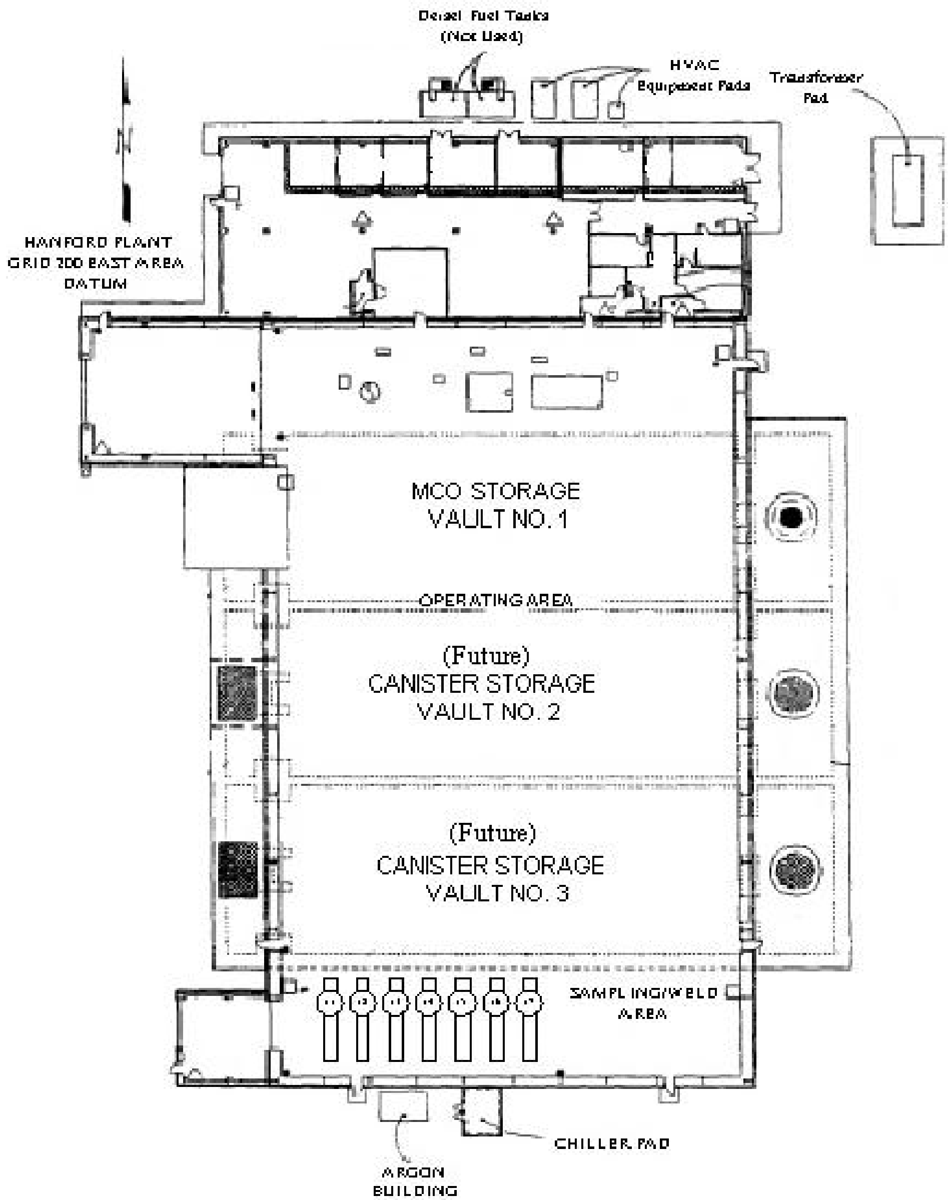

Figure 2-8. Canister Storage Building: Floor Plan

Source: Fluor Hanford, Inc., HNF-3553, Annex A, Revision 0 (Fluor Hanford, Inc., 2000)

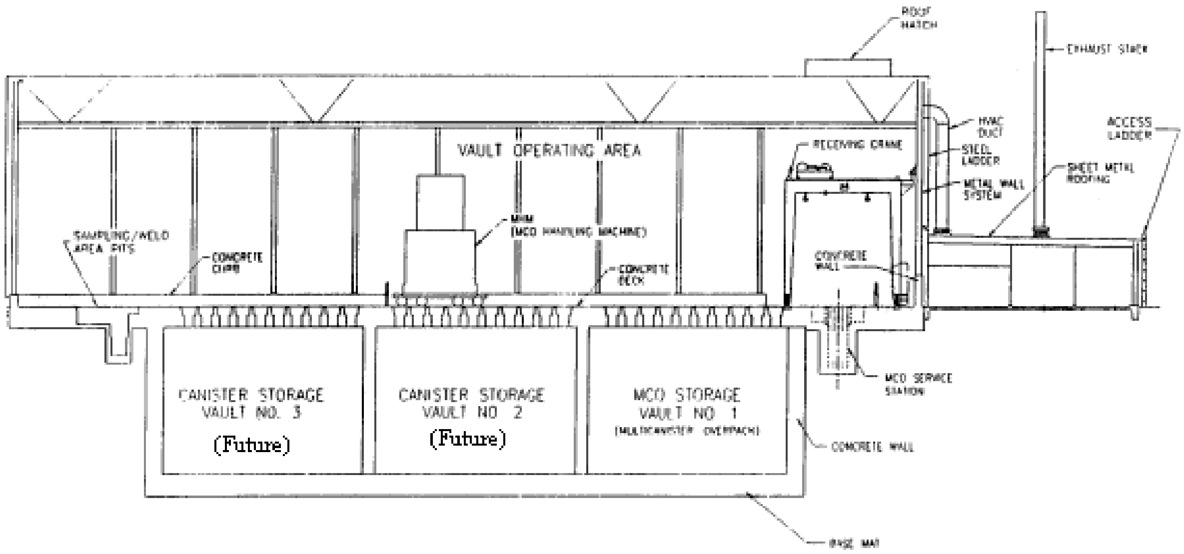

Figure 2-9. Canister Storage Building: Section

Source: Fluor Hanford, Inc., HNF-3553, Annex A, Revision 0 (Fluor Hanford, Inc., 2000).

Once the MCO transfer cask reaches the CSB, the MCO is removed from the cask by the MCO Handling Machine (MHM), transported to a storage location, and lowered into a storage tube. The MHM is a shielded rotating turret mounted on a bridge and trolley (see Figures 2-10 and 2-11). The turret has three cavities—one for moving the MCO, one for removing the storage tube plug, and one for location determination. The MHM is also used for moving each MCO to a sampling/weld station for installation of a welded cover over the shield plug and mechanical seal. The MCO is then returned to the storage tube for interim storage pending ultimate disposal in a repository.

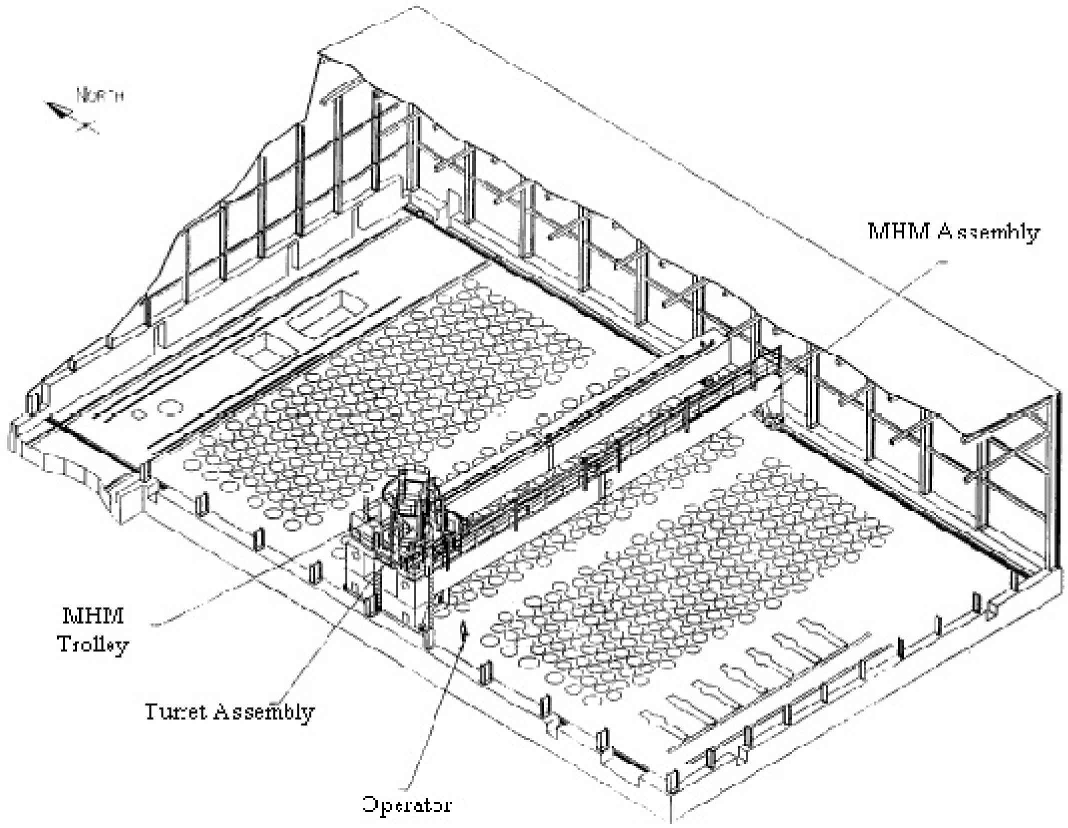

Figure 2-10. General Arrangement of Multi-canister Overpack Handling Machine

Source: Fluor Hanford, Inc., HNF-3553, Annex A, Revision 0 (Fluor Hanford, Inc., 2000).

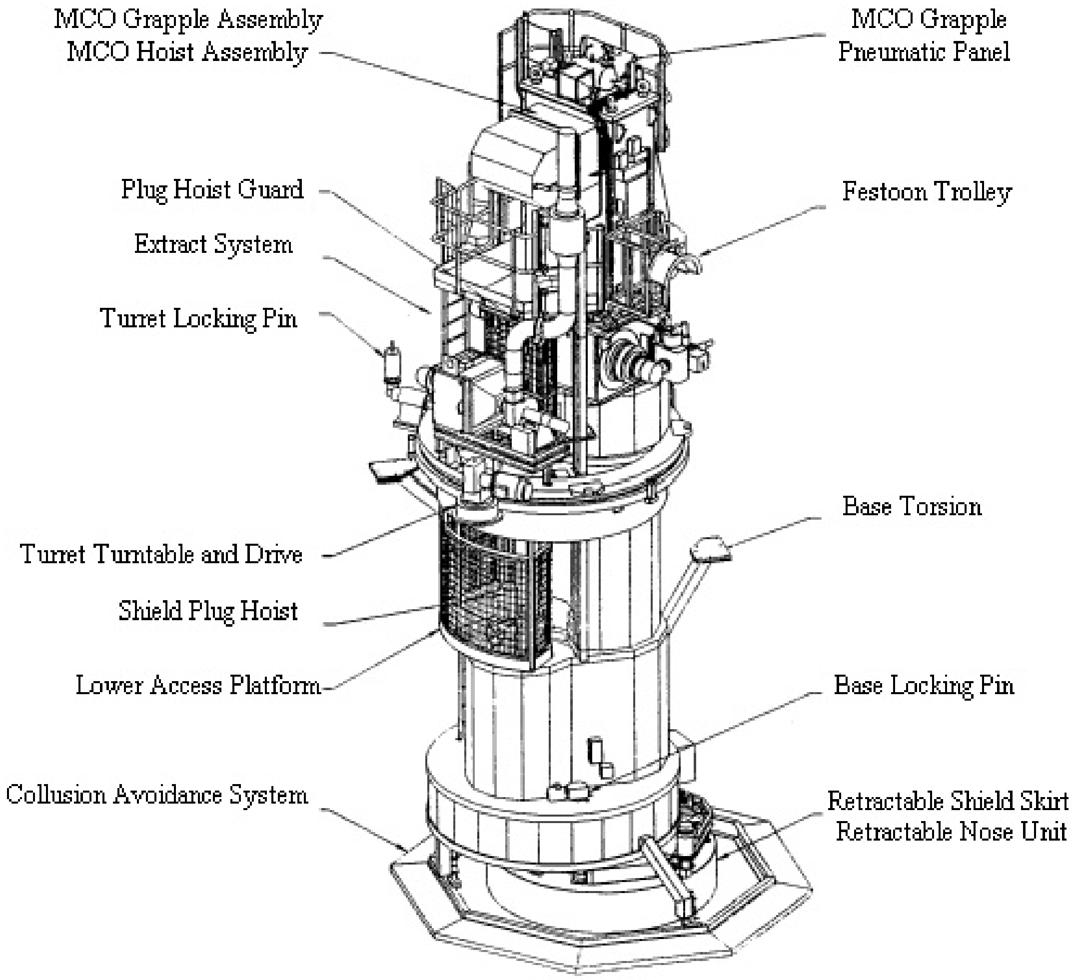

Figure 2-11. Cask Turret Assembly of Multi-canister Overpack Handling Machine

Source: Fluor Hanford, Inc., HNF-3553, Annex A, Revision 0 (Fluor Hanford, Inc., 2000).

2.3 MODELING OF PROCESSES AND FACILITY OPERATIONS

The success of the SNFP mission will be measured by how expeditiously all hazardous material is removed from the K-Basins and safely stored. Timely mitigation of the hazards in the K-Basins will reduce the safety risk to the public, workers, and the environment. To evaluate the effectiveness of SNFP processes and facility operations, the WITNESS™ model was used. WITNESS™ is a discrete simulation software tool commonly used in the commercial industry to determine capability, capacity, efficiency, and utilization of equipment, as well as to investigate system queues, bottlenecks, and other parameters. This tool was applied to assess baseline changes in the overall time to complete fuel removal operations from both K-Basins. These latest model runs reflect DOE's desire to store sludge from fuel removal operations in the T-Plant instead of treating the sludge directly from IWTS vessel storage (Cleveland and Pajunen, 2000).

The usefulness of this type of model depends on the availability of accurate operational data describing the characteristics of each subsystem (e.g., operator fuel sorting efficiency, IWTS equipment availability, MCO processing times). Since these data are unavailable for the SNFP, process times were modeled using probability distributions to describe the longest time, shortest time, and median cycle time for each production activity. Review of these distributions showed them to be reasonable, given information available at the time the model was run.

The following conditions were applied in the development of the project completion date of August 31, 2004, for removal of all SNF and sludge from the K-W and K-E Basins.

- Cycle times reflect fiscal year 2000 operating estimates.
- Start of removal of fuel from the K-E Basin begins 21 months after the start of fuel removal from the K-W Basin. (Note: The project schedule now shows 25 months.)
- The number of CVDF process bays will be reduced from three to two since concurrent fuel removal from both basins is eliminated.
- The number of MCOs to be sampled will be six.
- SNF currently stored in the T-Plant will be removed, dried in the CVDF, and stored in the CSB.
- Operations will be conducted 3 shifts/day, 7 days/week, as necessary.

Model vulnerabilities were identified and include the potential compromise of any of the following assumptions:

- Labor resources are unlimited, available whenever and wherever they are needed.
- The assumed operating efficiency of nonredundant subsystem components is uniformly high, e.g., IWTS (70 percent), MHM (90 percent).
- Project funding is unlimited.
- Operating requirements remain constant over the life of the project (e.g., safeguards and security requirements and/or environmental regulations do not change).

Once the project has refined the above assumptions and other model inputs based on actual operational experience, a more likely prediction of overall time to remove fuel from each basin can be developed. This reassessment would provide a more realistic estimate of DOE's ability to realize the following schedule commitments to the Board, contained in the *Implementation Plan for the Remediation of Nuclear Materials (Revision 3)* (U.S. Department of Energy, 2000), for Recommendation 94-1 (Defense Nuclear Facilities Safety Board, 1994).

- Complete fuel removal from K-W Basin by December 31, 2002.
- Begin fuel removal from K-E Basin by December 31, 2002.
- Complete fuel removal from K-E Basin by July 31, 2004.
- Begin sludge removal from K-Basins by December 31, 2002.
- Complete sludge removal from K-Basins by August 31, 2004.

3. REVIEW OF DESIGN AND CONSTRUCTION PHASE ACTIVITIES

The Board's staff reviewed the SNFP design and construction activities, as well as the project safety documentation, focusing on those structures, systems, and components affecting the safe operation of the facilities. The Board's staff also evaluated the technical safety requirements and administrative controls established to ensure safe operation and provide defense in depth.

3.1 PROJECT APPROACH TO HAZARD ANALYSIS AND CONTROLS

The SNFP contractor performed a structured and systematic examination of the project facilities and associated support structures and systems using standard industry techniques for hazard evaluation. This hazard analysis includes identification of the hazards associated with the project design, processes, and operations, addressing material at risk that could have an adverse effect on people or the environment and potential energy sources that could contribute to the uncontrolled release of that material.

The radiological source term assumes that all the fuel contains a mix of isotopes corresponding to the Mark IV fuel discharged from the N-Reactor in February 1979, representing a plutonium-240 (Pu^{240}) content of 16.72 percent (ratio of Pu^{240} to total plutonium) and 6.9 metric tons of uranium. This amount of uranium constitutes only a small fraction of the total of approximately 2,100 metric tons currently being stored in the K-Basins. This assumption is very conservative because concentrations of dose-significant radionuclides from the worst 0.3 percent of the fuel are used to represent the entire spent nuclear fuel (SNF) inventory. The methodology for calculating toxicological releases involves the release of SNF containing fission products and uranium, plutonium, and americium. The consequences from chemically hazardous materials are adequately controlled by the controls imposed to limit radiological releases. Hazardous conditions are assessed to determine a qualitative frequency of occurrence ranging from anticipated to beyond extremely unlikely. Potential impacts on the health and safety of the public and workers and on the environment range from those having unacceptable consequences for the off-site public to those having no impact.

In the initial accident screening, all hazardous conditions not having the potential to exceed the radiological consequences identified in Table 3-1 were deemed to require no further evaluation. This is consistent with applicable DOE guidance and is more conservative than current commercial practice. For each hazard type (e.g., load drops, gaseous releases), the most severe accident in terms of consequences and the highest-risk accident in terms of likelihood and consequences were selected for analysis. In some cases, the highest-risk and most severe accidents coincide.

The objectives of these analyses were to (1) identify the necessary and sufficient SSCs and Technical Safety Requirements that would result in satisfying the release limit and evaluation guidelines, and (2) provide sufficient defense in depth. The radiological release limits and evaluation limits and guidelines for the off-site public and on-site workers (collocated workers within 100 meters of the source) recommended by DOE Richland Operations Office (DOE-RL) are summarized in Table 3-1. Safety-class SSCs were identified to prevent or mitigate releases to the public that would otherwise exceed the off-site radiological limits or to prevent accidental criticality. Safety-class SSCs were also selected to meet the criteria of DOE Order 6430.1A, *General Design Criteria*, Section 1300-3.2 (U.S. Department of Energy, 1989).

Table 3-1. Radiological Evaluation Limits and Guidelines

Event Frequency Category	Frequency Range (per year)	On-site Risk Evaluation Guidelines (roentgen equivalent man [rem])*	Off-site Accident Release Limits (rem)*
Anticipated	10^{-2} to 10^{-1}	1	0.5
Unlikely	10^{-4} to 10^{-2}	10	5
Extremely Unlikely	10^{-6} to 10^{-4}	25	5

* All doses are cumulative effective dose equivalent.

Safety-significant SSCs were selected to prevent or mitigate releases of radioactive materials or toxic chemicals to collocated on-site workers. These SSCs include barriers that are judged to contribute substantially to defense in depth, independent of quantitative analyses. Safety- significant SSCs also protect facility workers from serious injury resulting from hazards not controlled by institutional safety programs.

The SNFP approach to hazard analysis and controls is based on the methodology recommended by DOE-STD 3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports* (U.S. Department of Energy, 1994). The SNFP criteria for system classification are also consistent with evaluation guidelines presented in Appendix A to DOE-STD-3009-94 and accepted by the Board in its July 8, 1999 letter (Conway, 1999).

3.2 K-BASINS

3.2.1 Definition of Functions and Requirements

The primary safety functions of the K-Basins are to provide radiation shielding and to remove heat generated by decay of the fission products in the SNF while operations for cleaning, packaging, and loading of SNF into shielded containers for transfer to the CVDF are conducted.

The removal of fuel from the K-Basins requires that the fuel be properly packaged and loaded into a transportation system. In this process, fuel in existing canisters (14 elements per canister) is moved to a fuel retrieval location within the basin. The fuel is cleaned using a tumbling action in a cleaning machine to remove sludge and corrosion products. Clean fuel elements and fuel pieces larger than 3 inches in length are loaded into fuel baskets. Normally, five or six loaded fuel baskets containing 270 or 288 fuel elements, respectively, depending on the type of fuel involved, are inserted into an MCO. Fuel debris (pieces larger than 1/4 inch and smaller than 3 inches in diameter) is placed into scrap baskets that are also loaded into an MCO in place of a fuel basket (see Section 3.3 below). The CLS (see Sections 2.2.1 and 3.2.3.3) is used to load the fuel and scrap baskets into an MCO contained in a transfer cask submerged in the basin. The loaded cask is then removed from the basin and placed on a transport vehicle for movement to the CVDF.

3.2.2 Analysis of Hazards

The Board's staff notes that the extensive cleaning, handling, and repackaging of the SNF resulted from DOE's desire to reduce the volume and number of containers to be stored in the CSB. There is also a legitimate desire to segregate small particulate matter, including broken pieces of fuel elements, from the other material in an MCO. This approach enhances the ability to dry the fuel in the CVDF, and establishes more confidence in the ability to provide ultimate disposal of the SNF in an approved high-level waste repository. The sludge and fuel particles removed by the IWTS need to be processed for ultimate disposal. A sludge treatment subproject currently in progress as part of the SNFP is aimed at developing the best method for retrieval, transportation, and storage of this material in the T-Plant facility. Ultimate processing and disposal will be determined as part of a site-wide initiative for disposal of like materials.

The most significant hazards in the K-Basins result from any of the following postulated events:

- Inadvertent criticality event.
- Runaway thermal reaction of uranium with water, leading to airborne release of radioactive material and hydrogen.
- Release of contaminated basin water to the environment, resulting in exposure of SNF and sludge, leading to airborne release of radioactive material.

3.2.2.1 Criticality Hazards

The limit on the effective multiplication factor (k_{eff}) for operations in the K-Basins is 0.98 (and 0.95 for subsequent activities involving the MCO in the CVDF and the CSB). The Board's staff considers these limits acceptable. All credible contingencies have been evaluated and appropriate criticality safety controls established. Designation of safety-class features to prevent criticality in the K-Basins is discussed in Sections 3.2.3.1 and 3.2.3.2.

In essentially all of the criticality events hypothesized, an assumption of multiple failures is necessary to reach a k_{eff} of 0.90 to 0.93. Since all but one of these cases were found to be acceptable, with k_{eff} below 0.95, the input assumptions were not challenged. For the excepted case, a cask-drop accident, the k_{eff} increased to 0.96. The contractor and DOE-RL considered this value unrealistic because of the conservatism of the assumptions in the analysis, including the following:

- The shipping cask breaches.
- The MCO breaches.
- The breach of the MCO causes a release of hydrogen that exceeds the lower flammability limit.
- An ignition source is present.
- The hydrogen ignites in a fire that requires the Hanford Fire Department to respond.
- The geometry control of the Mark IA basket center post and base plate fails.
- All the fuel collapses into rubble.
- The fuel and scrap rubble is optimally spaced.
- The resulting array is fully moderated by an unspecified source of water.

Only when all of these assumptions occur concurrently is the limiting value exceeded. If the rubble contains only fuel, the k_{eff} is less than 0.95.

Despite the conservatism associated with the generic scenario involving inadvertent criticality, a key element affecting criticality safety in K-Basin operations entails a large number of administrative controls. These controls are provided to ensure that mass limits on the contents of various materials in containers or baskets are not exceeded. Many of these activities, such as identification and weighing of various fissile material types and loading of SNF elements into the MCO baskets, demand the highest levels of performance by operating personnel. During an August 22, 2000, video conference between the Board's staff and representatives of both DOE-RL and SNFP, criticality safety personnel indicated that they had provided adequate training to operating personnel. The planned contractor Operational Readiness Review will ensure that the requirements from the Criticality Safety Evaluations (CSEs) are implemented in appropriate procedures and that personnel are adequately trained. In addition, SNFP representatives committed to the attendance of criticality safety engineers at prejob briefings for personnel from each operating shift and maintenance of an on-the-floor presence by criticality safety engineers to ensure that administrative procedures for criticality safety are followed.

Los Alamos National Laboratory recently detected and corrected approximately 40 errors in the maintenance of the Monte Carlo N-Particle (MCNP) code and developed a revised code, identified as MCNP4C. Since the earlier version of this code was used to prepare numerous SNFP CSEs, the Board's staff suggested review of the revised code before commencing fuel movement activities to confirm that the changes would have no significant effect on the K-Basin CSEs. During a site visit completed by the Board's staff on October 11–12, 2000, DOE-RL and the contractor indicated that such a review had been completed with insignificant effects on criticality margin. There were only minor differences in k_{eff} for a number of cases using MNCP4B versus MNCP4C.

The staff notes that the Criticality Safety Support Group (CSSG), established in accordance with the Board's Recommendation 97-2, *Continuation of Criticality Safety* (Defense Nuclear Facilities Safety Board, 1997), conducted an independent review of criticality safety for the MCO in August 1999. The CSSG's report concludes that the MCO and baskets are criticality safe as designed and do not require further modification. The CSSG review was conducted prior to the completion of a number of relevant CSEs. All relevant CSEs have since been completed and have been independently reviewed by an outside team, as well as by the Board's staff. The staff concludes that the MCO criticality analysis is adequate.

3.2.2.2 Hazards from a Runaway Thermal Reaction

As metallic uranium corrodes (oxidizes), it releases heat. Small fuel pieces in the scrap basket expose a large surface area subject to corrosion, thereby increasing the potential for a rise in local water temperature and thus an increase in the corrosion rate. A rise in the temperature also increases the potential for fuel cracking and crumbling, which in turn exposes new unoxidized fuel surfaces. If an adequate means of heat removal is not available, this process becomes divergent, leading to rapid corrosion of a large quantity of fuel and increasingly rapid release of large amounts of heat. This scenario is referred to as a runaway thermal reaction.

The French have experienced ignition of bare, irradiated, uranium fuel due to friction, leading to rapid oxidation and total consumption of bare uranium fuel cylinders in cold water. Because the Primary Cleaning Machine (PCM) tumbles defective fuel elements with areas of bare uranium exposed, the potential may exist for ignition and rapid oxidation that could lead to a runaway thermal reaction in the PCM. There has been no prior experience with the deliberate tumbling of irradiated defective fuel elements. The contractor has reviewed the French experience and has concluded that such a reaction is incredible for N-Reactor fuel in the basin water at a temperature of 10° centigrade (C).

The knockout pot in the IWTS collects fine particles of uranium, creating the possibility of rapid oxidation leading to a runaway thermal reaction. The contractor analyzed this potential problem and took action to reduce the possibility of its occurrence by designing copper cooling ribs on the interior of the knockout pot. With this extra cooling, the contractor has concluded that a runaway thermal reaction in the knockout pot is incredible. The potential for runaway thermal reactions also exists when small pieces of fuel are loaded into the scrap baskets, when baskets are loaded into the MCO and until the water is removed from the MCO in the CVDF. The contractor has analyzed these scenarios and judged them to be incredible.

A video conference was held on August 24, 2000, between the Board's staff and SNFP representatives to address the Board's staff concerns with runaway thermal reactions arising from rapid uranium oxidation that might occur in the K-Basins and in the MCO prior to removal of water. The Board's staff noted that although the contractor's analytical calculations indicated that runaway reactions were incredible, the models used in these calculations had not been verified by prototypical experiments or production experience and yield only analytical estimates. In addition, the SNFP cleaning and drying operations include processes never before used on damaged uranium fuel elements (see Section 3.3.2 for further discussion of hazards associated with the MCO during its transport to the CVDF, thence to the CSB, and during interim storage).

The Board's staff noted that although exposures to the public and collocated workers as a result of an unmitigated runaway thermal reaction within the K-W Basin were within guideline limits, thus eliminating the requirement for safety-class or safety-significant systems, it would be prudent to identify defense-in-depth mitigative systems and procedures. If a runaway thermal reaction occurred, facility workers could receive very high doses, and work areas could be significantly contaminated, resulting in delays in this important risk reduction project. The contractor and DOE-RL agreed to identify defense-in-depth design features and procedural controls to prevent or mitigate potential runaway thermal reactions.

An issue report prepared by the staff and forwarded by a letter from the Board dated September 20, 2000 (Conway, 2000) suggested several additional design features and procedural controls to prevent and mitigate runaway thermal reactions for evaluation by the project. These issues were resolved by the Board's staff with representatives of DOE-RL and the contractor during a site visit on October 11–12, 2000, as noted below.

K-Basin Fuel Removal Operation:

- The Board's staff had earlier identified a need for increased control of loading in the original knockout pots, which have limited heat transfer capability. In response, as approved by DOE-RL in July 2000, the design of the knockout pots has been changed to include copper cooling surfaces. This change will improve heat conduction and provide increased margin against a runaway reaction. Replacement pots will be procured and installed after initiation of fuel removal operations. The knockout pots currently installed in the basin do not have copper cooling surfaces and are now subject to a loading limit, which has been included in the appropriate operating procedure.

- The staff had also identified a need for K-Basin operators to recognize a potential runaway thermal reaction and to take appropriate action to protect workers. The contractor subsequently has identified three additional steps to protect workers during K-W Basin operations. First, formal familiarization of the K-W Basin operators has been provided addressing the corrosion behavior of uranium metal, the French experience, characteristics of a hypothetical runaway thermal reaction; and prudent measures to be taken to minimize risks. Second, additional Continuous Air Monitor alarms will be placed in work locations where potential runaway thermal reactions could occur in order to give workers the earliest possible warning of a problem. Finally, an emergency action plan will be developed to address recovery of operations in the K-W Basin in the event that operations are suspended due to a runaway thermal reaction.

Transportation of the MCO/cask from the K-W Basin to the CVDF.

! The staff identified a need for a requirement to measure the pressure of the MCO/cask head space if the transfer from the K-Basins to the CVDF exceeds 24 hours, because excessive pressure could indicate the beginning of a runaway thermal reaction. The transportation procedure has been clarified and now requires the MCO/cask head space pressure to be measured if the transit time exceeds 24 hours. If excessive pressure is measured, corrective actions have been identified.

! The staff also identified a need for a requirement to measure the pressure of the MCO/cask head space upon receipt at the CVDF, even if the transit time is less than 24 hours, to verify that there has been no significant increase in pressure. The operating procedure now requires pressure to be measured upon receipt of the MCO/cask. Corrective actions are required in the event excessive pressure is measured.

3.2.2.3 Hazards from Loss of Basin Water and Exposure of Fuel

The existing authorization basis for the K-Basins addresses the potential for release of water to the soil and for uncovering of the fuel and sludge, which could become airborne upon drying out. Leakage from the basin is limited, and sources of makeup water have been identified. The authorization basis includes installation and maintenance of isolation barriers to separate the main basin from a construction joint that is vulnerable to leakage. These isolation barriers were installed in 1995 to satisfy a commitment in the Implementation Plan for Recommendation 94-1 (O'Leary, 1995) .

An additional hazard in this area was introduced as a result of the SNFP modifications. A drop of a loaded MCO inside a transfer cask in the South Loadout Pit could result in significant damage to the floor-to-wall joint. This hazard was reviewed extensively based on classification of the basin walls and floor as safety-class structures, requiring that leakage from the basin be limited. This issue has been addressed by operating controls and a leakage mitigation system (see Section 3.2.3.3.).

3.2.3 Identification of Hazard Controls

3.2.3.1 Fuel Retrieval System

The FRS, described in Section 2.2.1, has a limited number of safety-class components. The K-Basin Final Safety Analysis Report (FSAR) designates four FRS SSCs as safety-class equipment:

- PCM lower half and support structure.
- Process table support structure and MCO basket “go-no-go” gauges and bottom plates.
- MCO basket queue.
- Tether system for manipulator rail support structure.

The PCM support structure and attachment bolts are safety-class to ensure that the PCM will not collapse or tip over because of a load drop or seismic event. The process table support structure ensures that the fuel contained in the MCO basket will not spill onto other fuel, potentially causing a criticality. The MCO basket queue is also designed to prevent the fuel in the MCO baskets from spilling out. The manipulator rail support structure is tethered to prevent it from falling into the basin if the support structure should fail because of overstress resulting from a design basis earthquake.

In the enclosure to a letter dated July 8, 1999 (Conway, 1999), the Board’s staff addressed a design failure involving the PCM. The PCM cleans the SNF by mechanical agitation and a water wash before the SNF is sorted and loaded into fuel and scrap baskets. The central part of the PCM is a stainless steel screen drum that is split axially into two halves. The drum is oriented horizontally and rotates about its axis, causing a tumbling action for the SNF canister loaded inside. Failures occurred in the split-shaft design during factory acceptance testing. The original design requirements dictated that operators be able to lift either half of the drum from the PCM base to empty and remove the canister in either the upright or the inverted position. The tested design failed as a result of excessive wear and galling of the bearing surfaces. The Board’s staff questioned the quality of the original design effort and the level of independent review. An independent team was subsequently assembled to analyze the original design and recommend modifications that would resolve the problem. The team reviewed the original design and developed a modified split-bearing design, which was installed and tested successfully.

3.2.3.2 Integrated Water Treatment System

The IWTS, described in Section 2.2.1, has the following safety-related components:

- Inlet strainer screen for the sludge pumping system.
- Knockout pots.
- Knockout pot screens.
- Knockout pot lifting hook.

- Particulate settler vessels.
- Annular filter vessels.

The inlet strainer screen for the sludge pumping system prevents particles greater than ¼ inch from entering the IWTS, thereby preventing inadvertent criticality in the knockout pot.

The knockout pot is designed to limit the volume and geometry of SNF to (1) prevent inadvertent criticality, and (2) provide sufficient assurance that after a load drop or seismic event, SNF will remain contained within the pot. The knockout pot screen is designed to prevent inadvertent criticality in downstream vessels by limiting the particle sizes capable of passing through the screen (#500 microns nominal). A project study determined that an insignificant amount of screen erosion is expected during the limited operational life of each knockout pot. The knockout pot lifting hook limits the drop height to ensure that knockout pots will contain fuel during and after an inadvertent drop.

Particulate settler vessels are also designed to limit the volume and geometry of SNF, thereby preventing inadvertent criticality by controlling vessel dimensions. The annular filter vessels prevent criticality as well, by limiting the SNF volume and controlling its geometry. These vessels have a tank-in-tank design, with the inner tank normally being empty. The annular space between the tanks contains approximately 90 cubic feet of sand/garnet filter medium.

All vessels within the IWTS represent passive safety features and therefore have no associated Technical Safety Requirements (TSRs). Each vessel was designed and constructed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section VIII (American Society of Mechanical Engineers, 1998). The knockout pot screens and the inlet strainer screen for the sludge pumping system were built to ASME B31.1, Power Piping Code.

3.2.3.3 Cask Loadout System

As noted earlier, it was determined in early 1999 that a postulated drop of a loaded MCO transfer cask could cause failure of the wall-to-floor joint of the South Loadout Pit. The basin floor and walls are designated as safety-class structures. Use of the original design could have resulted in excessive basin leakage had a cask drop of more than 1 foot occurred. Redesign of the original CLS was proposed to address this issue. Alternative loadout systems were also reviewed. At that time, the Board's staff urged DOE-RL to provide adequate justification for eliminating several of the alternatives to the original design approach and to place more emphasis on prevention of the postulated cask drop. DOE-RL decided to require the design of a new support structure incorporating a nesting pail to provide hydraulic damping, thus limiting the floor impact.

The Board's staff reviewed the proposed redesign of the CLS to dissipate the energy from a cask drop and limit the impact on the floor. Review comments were provided in an issue report (Grover, Wille, 1999) forwarded by a letter from the Board dated July 8, 1999

(Conway, 1999). As a result of this letter and continued design evaluations, the contractor revised the design to eliminate the impact on the floor and the potential for damage to the floor-to-wall joint.

In a letter to DOE-RL dated July 29, 1999 (Williams, 1999), the contractor recommended returning to the original CLS design instead of proceeding with the proposed new design. DOE-RL accepted this recommendation and noted that the project would (1) obtain and implement guidance from the Navy Crane Center on minimizing the probability of a cask drop; (2) obtain and pre-stage sealant injection equipment to mitigate potential basin leaks in the unlikely event of a drop; (3) develop procedures for and train personnel in use of that equipment; and (4) install a maximum-thickness crushable pad below the CLS in the South Loadout Pit.

In light of the potential for a cask drop accident, the Board's staff urged that the project evaluate risk-reduction methods, especially after the decision was made to take a risk-based approach by using the original CLS design for both basins. Emphasizing the prevention of a cask drop, the Board's staff performed a comprehensive review of the MCO transfer cask handling system. The K-W Basin has a transfer bay crane to be used for cask handling. The crane is rated at 32 tons and load tested at 40 tons (125 percent of rated). The weight of the loaded MCO transfer cask is calculated to be approximately 29.5 tons; hence the crane will have adequate margin for cask handling operations. Because of the high number (~800) of lifts at near capacity, however, the staff gave special attention to each of the crane's safety features.

As part of the upgrade to the current rated capacity of 32 tons, numerous safety features were incorporated. These features included an additional set of hoist brakes, misreeving protection, load cells, and a second independent set of upper limit switches aimed at reducing wire rope failures (which account for most dropped loads). However, human errors are responsible for nearly all of the initiating events leading to wire rope failures and other causes of dropped loads. Therefore, the Board's staff urged an assessment of proposed controls for operation and maintenance of the cranes, including provisions for crane inspection and operator training.

The Navy Crane Center conducted an assessment of hoisting and rigging for the SNFP during the week of May 22, 2000. Center representatives stated that they consider the K-W Basin 32-ton bridge crane safe, but noted several deficiencies that could affect reliable service during critical lifts. The project developed a plan to implement the recommendations. However, the Board's staff considered the implementation to be untimely, since the planned Operational Readiness Review (ORR) lifting demonstrations with a loaded dummy MCO/cask represented a risk of a wall-to-floor joint failure equal to that posed by the post-ORR lifts of an MCO/cask loaded with radioactive fuel. The following issues were identified in a Board letter (Conway, 2000) and were resolved by the Board's staff with the project during a site visit on October 11–12, 2000:

- The Board's staff identified a need to retest the 32-ton crane in the K-W Basin to its rated capacity. A load test of this crane had been performed in November 1999, when repairs were made to the main hoist electric brake. This load test was done using 24 tons, which is only approximately 80 percent of the weight of a loaded MCO/cask. The K-W Basin 32-ton bridge crane was subsequently load tested at its rated capacity of 32 tons, following upgrades of the programmable logic controller.

! The staff also identified a need to provide timely exercise of the same crane, which had a history of electrical faults and trips since it was redesigned. Following the upgrades and load test discussed above, the K-W Basin 32-ton bridge crane was extensively exercised to verify reliability.

! The staff identified a need to demonstrate the ability to seal basin leaks prior to lifting a loaded dummy MCO/Cask in the K-W Basin. The sealant injection equipment designed to mitigate potential basin leaks in the unlikely event of a cask drop, including appropriate procedures, training, and drills in the use of that equipment was scheduled for completion prior to the start of production operations. A drill demonstrating the use of the sealant injection system was conducted prior to the ORR lifting demonstration.

3.2.3.4 Electrical and Instrumentation and Control Systems

The K-Basin Safety Analysis Report (SAR) classifies the electrical and instrumentation and control systems as non-safety-related and describes the electrical distribution system for both the K-E and K-W Basins. Three-phase power is supplied from a commercial source at 230 kilovolts (kV) delivered to a radial distribution system. A transformer steps down the voltage to 13.8 kV at the on-site substation, and a bus cross-tie connects the K-E and K-W Basins. Voltage is further reduced as necessary at the K-Basins for 4,160 volt (V) and 480 V distribution.

The system is designed with protective relaying for the 13.8 kV and 480 V electrical distribution systems. The 480 V system is also protected by solid-state trips. Motor control center loads are protected individually by molded-case circuit breakers. Single-phase distribution of 120 V is also available via lighting panels. The 125 V, direct current power system used for switchgear controls consists of 60 lead-acid cells and a charger.

Comprehensive short-circuit, voltage profile, and coordination studies are essential to safeguard personnel and maintain a safe and reliable power system. These studies should be performed in accordance with appropriate Institute of Electrical and Electronics Engineers (IEEE) standards.[1] According to the K-Basin design requirements document, electrical system components must be coordinated for short-circuit capability, interrupting duty and capability, insulation levels, protective relaying, reliability, interchangeability, transformer and line voltage drop, stability under normal conditions, and restart upon power dips and outages. The Board's staff confirmed that these calculations and studies have been performed.

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[1] IEEE-141, *IEEE Recommended Practice for Electrical Power Distribution for Industrial Plants,* and IEEE-242, *IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems*.

In the enclosure (Gwal, 1998) to a letter dated February 25, 1998 (Conway, 1998), the Board's staff noted that existing battery rooms at the K-Basins did not comply with applicable codes and standards relating to battery room ventilation systems.[2] These rooms had no detection systems for loss of exhaust ventilation. During a tour of one of the battery rooms, the staff observed a broken belt on an exhaust fan motor, leading to loss of exhaust ventilation. Under these conditions, an explosion could result if accumulated hydrogen concentration exceeded the 4 percent lower flammability limit and came in contact with an electrical spark from battery cabling or connections. To resolve this issue, an in-operation annunciator was installed, and surveillance requirements were established to alert the shift operator in the event of a loss of battery room ventilation.

In the enclosure to the same February 1998 letter from the Board, the staff addressed the issue of the calibration of the electrical switchgear protective relays in the K-W Basin facility. The Board's staff identified a need to verify the adequacy of these calibrations in an issue report forwarded by a letter from the Board (Conway, 2000). The issues, which are related to electrical switchgear at several voltage levels, were resolved by the Board's staff with the project during a site visit on October 11-12, 2000, as follows:

- The original issue identified by the staff concerned 480V circuit breakers. The solid-state overload relays protecting three 480V circuit breakers were modified and recalibrated during installation of new systems completed in November 1999. Calibration of the remaining nine solid-state relays serving this function is scheduled to be checked in late 2000 and early 2001.

- The Board's staff had also raised the issue of maintaining current calibration of 4,160V protection relays. SNFP representatives stated that about half of the approximately 60 relays in this category have been checked for calibration and recalibrated, as required. Calibration checks on the remainder are being conducted on a contingency basis. The Board's staff finds this approach acceptable.

- During the October 2000 site visit cited above, the Board's staff questioned the status of calibration of protection relays for the 13.8 KV electrical system. SNFP representatives determined that the contractor responsible for maintenance of this switchgear has completed calibration of these relays and that their calibration is current.

3.2.3.5 Fire Protection

[2] American National Standards Institute (ANSI) C2, *National Electrical Safety Code*; National Fire Protection Association (NFPA)-70, *National Electrical Code*; and IEEE Std 450-1995, *IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries for Stationary Applications.*

The K-Basin fire protection system consists of radio fire alarm reporting boxes, fire alarm control panels, and a water supply system for each of the sprinkler systems. Fire protection systems are required primarily for property protection in the K-Basin area. Combustible controls limit permitted quantities of combustibles, as well as separation distances from structural elements. There is a local fire protection program, as well as a mature, well-documented site-wide program. The fire hazard analysis (FHA) pertaining to the K-Basins meets the requirements of DOE Order 5480.7A, *Fire Protection*, (U.S. Department of Energy, 1993).

3.3 MULTI-CANISTER OVERPACK

3.3.1 Definition of Functions and Requirements

The safety function of the MCO is to provide containment of the SNF during stabilization and storage operations. A rupture disc in the shield plug assembly limits internal MCO pressure during processing. After processing, welding of the cover assembly to the MCO, which is done in the CSB, encloses the rupture disk and access ports, providing a totally welded, sealed container.

Pressurization of the MCO above design pressure during interim storage could lead to failure of the containment boundary and release of radioactive material. Potential sources of pressurization in the MCO during storage are the production of helium from alpha decay, noble gases resulting from the fission process, and radiolysis of water. After drying in the CVDF, essentially all the free water has been removed from the MCO. This is confirmed by a pressure rebound test in which the MCO is isolated after it has been evacuated, and the rate of pressure rise is measured. Any remaining free water in the MCO would cause the pressure to rise to the equilibrium pressure at the MCO temperature. Absence of a significant pressure rise confirms that essentially all the free water has been removed. Because the contributions from helium and fission gases are trivial, radiolysis of water of hydration is the only remaining pressurization process of concern.

Water of hydration in the sludge and oxide films on the fuel elements is not removed by the drying process. Much higher temperatures are required to decompose such chemically bound water. The amount of hydrated water in an MCO is estimated from the amount of sludge and adherent oxides on the fuel surfaces.

The project FSAR indicates that the expected MCO pressure during interim storage will be less than 16 psi(g). Using extremely conservative bounding values for sludge and adherent oxides gives a safety case bounding pressure of about 62 psi(g), all from released hydrogen (assuming that the released oxygen is absorbed by the bare uranium fuel).

The cover assembly is not in place during processing in the CVDF. During this time, a 150 psi(g) rupture disk in the shield plug protects the MCO from overpressurization. On the basis of the bounding analysis and to provide additional safety margin, the design pressure of the MCO with its welded cover assembly in place was set a factor of three above the pressure at which the rupture disc is designed to fail (i.e., at 450 psi[g]).

The Board's staff concurs with the dual design pressure for the MCO and agrees that the pressure increase will likely be less than that of the bounding analysis.

3.3.2 Analysis of Hazards

As described in the MCO Topical Report, upset events can occur at various MCO processing stages: (1) in the K-Basins, (2) during transport from the K-Basins to the CVDF, (3) during dewatering and drying in the CVDF, (4) during transport from the CVDF to the CSB, (5) during processing in the CSB, and (6) during interim storage in the CSB. Potential hazards at each of these stages are discussed below. Because its functions are to contain the spent fuel and to prevent a release of radioactive material to the environment, the MCO is designated as safety-class.

3.3.2.1 Hazards in K-Basins

The most significant hazard associated with the MCO while it is in the K-Basin is that of a runaway thermal reaction. To guard against such a reaction, the scrap baskets were modified to have radial copper ribs to enhance cooling. Thermal analyses showed that the modified scrap baskets have adequate margin against a runaway thermal reaction, and the event has been determined to be incredible (i.e., it has a probability of occurrence of less than 1 in 10^6). However, as a defense-in-depth feature, operator training and administrative controls will address runaway thermal reactions and appropriate operator response (see Section 3.2.2.2).

3.3.2.2 Hazards During Transport from K-Basins to CVDF

During transport to the CVDF, the MCO is in the shielded transfer cask. The MCO is vented to the interior of the cask, and the cask is sealed. Analysis of the potential for pressure buildup in the cask due to oxidation of the fuel showed that if the shipment were completed in less than 24 hours, no significant pressure buildup would occur. This analysis has not been verified by prototypical experiments or production experience. If there is no indication of buildup of gas from fuel oxidation before the shield plug is installed, it is reasonable to conclude that no extensive oxidation has taken place and that a runaway thermal reaction is unlikely to occur during the 24-hour transportation period (see Section 3.2.2.2).

The contractor's SAR indicates that if the transfer of the MCO to the CVDF is not completed within the specified 24-hour period, the MCO transfer cask should be vented to prevent any possible buildup of internal pressure. As the most likely source of an increase in internal cask pressure is heating of the fuel from its oxidation, and as no limits are set on either an acceptable pressure increase or the time period during which venting of the MCO transfer cask will be permitted, there is the possibility of a runaway thermal reaction during the venting period. As a defense-in-depth feature, operator training and administrative controls will address the potential for runaway thermal reactions and appropriate operator response (see Section 3.2.2.2).

3.3.2.3 Hazards During Dewatering and Drying in CVDF

The purpose of the CVDF operations is to remove the bulk water from the MCO, dry the fuel, refill the MCO with helium, and seal it. After the loaded MCO transfer cask arrives at the CVDF and before the cask lid is removed, the cask head space is vented (primarily for release of hydrogen). The cask is heated to 50°C to heat the fuel, then pressurized with helium to remove bulk water through the center dip tube, which extends to the bottom of the MCO. Vacuum drying of the heated SNF removes remaining free water from the MCO, minimizing the possibility of a runaway thermal reaction.

During a video conference on August 24, 2000, between the Board's staff and representatives of DOE-RL and the contractor, the staff was informed that internal pressure would be checked upon the cask's arrival at the CVDF, regardless of how long it had been in transit, to ensure that no significant increase in pressure had occurred. In the event of a significant pressure increase, which could indicate an incipient runaway thermal reaction, preventive or mitigative procedures would be followed. This defense-in-depth feature will include operator training and administrative controls to address appropriate operator response (see Section 3.2.2.2).

3.3.2.4 Hazards During Transport from CVDF to CSB

Once the fuel has been dried, the MCO is backfilled with helium and sealed. The water in the annulus between the MCO and the cask is removed, the annulus is backfilled with helium, and the MCO transfer cask is moved to the CSB. An administrative limit requires transport from the CVDF to the CSB within 135 days to avoid flammable gas buildup in the cask annulus, as described in the MCO Topical Report.

During transport to the CSB, the MCO is contained within and protected by the transfer cask, which is designed to withstand postulated accidents while in transit. This aspect of on-site transportation of the spent nuclear fuel is discussed in Section 3.4.

3.3.2.5 Hazards During Operations in CSB

Upon arrival at the CSB, the MCO transfer cask is moved into a service area where the pressure under the cask shield plug is checked. If the pressure has not increased, the cask shield plug is removed. The MCO is then removed from the cask using the MCO Handling Machine (MHM) and transferred to the CSB weld/test station, where pressure within the MCO is measured. If no significant increase in the MCO's internal pressure has occurred, the cover assembly is welded on the MCO. The result is a totally welded container, which is then placed in a storage tube for interim storage. Initially, six MCOs will be placed in interim storage in storage tubes without a welded cover assembly, for up to 2 years. Periodically during this surveillance period, the internal MCO pressure will be measured and the gas composition analyzed.

The following five major design basis accidents associated with operations in the CSB were evaluated by the contractor:

- Mechanical damage to an MCO.
- Gaseous release from an MCO.
- Internal hydrogen deflagration in an MCO.
- External hydrogen deflagration.
- Runaway thermal reactions inside the MCO.

Mechanical Damage to an MCO. An MCO or the cask-MCO combination could be mechanically damaged, resulting in a breach of the MCO. Such damage is postulated to result from a drop of the MCO, shearing of the MCO by the MHM, or other unspecified cause leading to a breach of the MCO.

A review of the lift and transfer paths for a loaded MCO transfer cask for all spent fuel operations determined that a cask drop in the CSB would produce the greatest acceleration. This bounding case, which could occur during movement of the cask from the cask trailer to the receiving pit, involves a 40-inch drop onto 5-foot-thick reinforced concrete that has a compressive strength of 9,000 pounds per square inch. Evaluation of this accident by the project indicates that criticality control and confinement design features remain intact.

Gaseous Release from an MCO. Gaseous releases from an MCO are postulated to arise from either overpressurization or leakage. The first condition involves the overpressurization (and subsequent breach) of an MCO by the inert gas system during reinerting of a monitored MCO after sampling. The second postulated condition involves a release of radioactive material from an MCO due to a leak caused by equipment associated with gas sampling or human error during sampling operations while an MCO is located at the sampling/weld station. MCOs arrive at the CSB in a pressurized condition because of helium backfilling operations performed at the CVDF (see Section 3.3.2.4).

MCO Internal Hydrogen Deflagration. The internal deflagration scenario postulates the ignition and burning of a hydrogen-oxygen mixture inside an MCO. In one scenario, oxygen is assumed to be introduced into the MCO when purge gas contaminated with oxygen is used to inert the MCO after sampling.

External Hydrogen Deflagration. An external hydrogen deflagration is postulated to occur outside an MCO. One such scenario involves the release of hydrogen from an MCO into the sample hood and exhaust system. After mixing with air, the hydrogen ignites and burns.

Runaway Thermal Reactions Inside an MCO. Two beyond-design-basis scenarios were postulated for this accident. The first involves the reaction of water with uranium fuel and uranium hydride, and includes the assumption that an MCO remains in the sampling/weld station for 40 days without active cooling. The second involves the reaction of oxygen with uranium fuel and uranium hydride, and includes the assumption that an MCO is inadvertently filled with oxygen at the sampling/weld station. The MCO temperature increases in both scenarios, but in neither does the breach of an MCO occur (see Section 3.3.1).

3.3.2.6 Hazards During Interim Fuel Storage in CSB

The dominant hazard to the MCO during interim storage is the loss of passive cooling, leading to possible violation of design temperature criteria (see Sections 3.6.2 and 3.6.3.2).

3.3.3 Identification of Hazard Controls

Because it is intended to provide containment during transportation and interim storage of the SNF, the MCO must be robust and reliable. The MCO was originally designed to requirements that essentially corresponded to those of the ASME BPVC, Section VIII (American Society of Mechanical Engineers, 1998). Reviews conducted by the Board's staff revealed that many significant design requirements were missing (Wille, 1998). Following numerous discussions involving the Board, its staff, DOE-RL, and the contractor, DOE-RL directed that the MCO be designed to the ASME BPVC, Section III, Division 1 (American Society of Mechanical Engineers, 1998), thus meeting the same requirements imposed on nuclear reactor pressure vessels. Consistent with the Board's urging during a site visit in August 1998, DOE-RL also decided that the MCOs would be code-stamped to provide enhanced quality assurance and to avoid future reassessments of the integrity of the MCOs. This set of requirements provides the highest design quality and highest manufacturing quality assurance available for pressure vessels manufactured by a commercial vendor. MCOs designed and manufactured to these requirements have been determined by DOE's Office of Civilian Radioactive Waste Management to be acceptable containers for disposal at the proposed radioactive waste repository at Yucca Mountain, Nevada.

The procurement contract was placed with a BPVC-certified vendor (Joseph Oat Co.) with a 30-year record of on-time delivery of many nuclear-grade BPVC components. In addition to the Authorized Nuclear Inspector required for code-stamped components, the SNFP contractor assigned a full time on-site inspector in the fabricator's shop to inspect the MCOs as the fabrication progressed. This combination of identifying the proper requirements, using a certified vendor, and providing on-site inspection at the fabricator's shop is resulting in on-time delivery of high-quality MCOs.

As noted in Section 3.3.2.4 above, the MCO is dried, backfilled with helium, and mechanically sealed, using a gasket, in the CVDF before it is transported to the CSB. A mechanically sealed and gasketed container does not meet the requirements of the ASME BPVC for storage of irradiated nuclear reactor fuel. To meet those requirements for the MCO, a 304L stainless steel cover assembly is welded to the top of the MCO, using a full-penetration field weld, thus providing a completely welded container. Because of the configuration and contents of a loaded MCO, this weld cannot be inspected radiographically to meet BPVC requirements. However, the ASME Boiler and Pressure Vessel Committee approved Code Case N-595, allowing dye penetrant inspection and leak rate testing as a substitute for radiography. This alternative approach will be used to confirm the integrity of the cover assembly weld.

To facilitate handling of the MCO, the exterior of the cover assembly is machined to have the same configuration as the locking ring (see Figure 2–4). The cover assembly has one mechanically sealed penetration that is aligned over a port on the shield plug to allow gas sampling of the atmosphere within the cover, as well as to permit access for operation of the shield plug port. The penetration is covered with a plate welded to the cover assembly.

During loading of MCOs in the K-Basins, records are kept of the condition of the fuel loaded into each MCO. From this information, as noted earlier, six sample MCOs representing worst-case conditions will be selected for a special monitoring program during their interim storage in the CSB. Under this program, pressure and temperature in these sample MCOs will be monitored continually, and analyses of the interior gas will be conducted quarterly for a 2-year period. Although this monitoring program is not part of the SNFP safety basis, these data will be used to confirm the prediction that no significant pressure buildup occurs during interim storage.

Although increases in MCO pressure during interim storage are not expected to be large, the Board's staff urged DOE-RL to provide an indication of the pressure buildup. For long-term monitoring, the contractor will include a pressure device in each MCO cover assembly to sense pressure under the cover. The device will employ magnetic coupling to transmit a signal from the pressure sensor inside the MCO to a readout device mounted on top of the cover, providing a gross indication of internal pressure before the MCO is handled. During an August 24, 2000, video conference, the contractor displayed the pressure sensor and readout gauge to be used. These devices will be field installed in each cover assembly by the SNFP contractor.

3.4 ON-SITE TRANSPORTATION OF SPENT NUCLEAR FUEL

3.4.1 Definition of Functions and Requirements

The primary safety functions of the Cask Transportation System (CTS) are to provide shielding from radiation emitted by the highly radioactive fuel elements inside a loaded MCO, and to provide protection and containment of the MCO during on-site movements. The system consists of a shielded MCO transfer cask and a tractor-trailer transporter capable of moving MCOs safely between facilities, and of serving as a temporary storage and handling device for MCOs during dewatering activities conducted at the CVDF (see Section 2.2.3). The functional requirements are leaktightness and sufficient physical strength to absorb energy and withstand drop loads and seismic forces.

The Safety Analysis Report for Packaging (SARP) provides a description of the MCO transfer cask and its operation as well as an analysis of the risks involved during normal transportation and accident conditions. As noted in the SARP, the design and structural analyses of the confinement boundary in the MCO transfer cask follow the criteria of the ASME BPVC (Section III, Subsection NB, Class 1). However, the cask is not code-stamped, in contrast to the code stamping of the MCO. (The MCO code stamp is necessary because of the MCO's use for interim storage in the CSB and ultimately for disposal in a high-level waste repository).

The DOE-RL Safety Evaluation Report (SER) for the SARP notes that, as documented in the SARP, the MCO cask fails to meet all of the requirements of Title 10 of the Code of Federal

Regulations (CFR), Part 71, *Packaging and Transportation of Radioactive Material* (National Archives and Records Administration, 1999), for off-site shipments; however, the SER further states that meeting such requirements is neither required nor necessary. The SER concludes that on-site shipments using the CTS present acceptable levels of risk. The Board's staff believes that the associated functions and requirements identified in the DOE-RL SER should be sufficient, recognizing the dedicated and restricted on-site use of the CTS, with robust MCOs.

3.4.2 Analysis of Hazards

3.4.2.1 Criticality Hazards

Evaluation of the risk of inadvertent criticality during transport of the loaded MCO transfer cask is provided in HNF-SD-SNF-CSER-010, Rev. 1B, *Criticality Safety Evaluation Report for Storage and Removal of Spent Nuclear Fuel from K Basin* (Kessler, 2000). The principal performance requirement for prevention of criticality under these conditions is to maintain k_{eff} at less than 0.95, as required by DOE's "nuclear safety equivalency" (comparable to Nuclear Regulatory Commission [NRC] requirements). The analysis shows that shipments of Mark IA and Mark IV MCOs will remain subcritical for all normal transfer conditions and for all credible accident conditions. The most severe hypothetical accident analyzed involves a drop of a flooded MCO loaded with 12 long-length (26 inches) Mark IA fuel assemblies in a single Mark IV MCO, causing the fuel to be broken into optimally sized and spaced rubble. This accident is considered incredible.

3.4.2.2 Release of Radioactive Material

The CTS will be used to make about 400 shipments of loaded MCOs from the K-Basins to the CSB, with an intermediate stop at the CVDF for drying of the SNF. This campaign is expected to take approximately 4.5 years. The hazard associated with this transportation is the potential release of radioactivity from the MCO during normal transportation and accident conditions. Although the MCO provides a containment barrier, the accident evaluations in the SARP take credit only for the MCO transfer cask. The SARP classifies the cask body and cask closure as critical elements in preventing the release of radioactivity. Both the K-Basin and CSB FSARs list the MCO transfer cask as safety-class equipment designed to protect the MCO from damage caused by load impacts, drops, or seismic events and to prevent accidental releases.

The MCO transfer cask design must meet external release rate requirements with a containment system that satisfies the leaktight criterion of ANSI N14.5, *Leakage Tests on Packages for Shipment* (American National Standards Institute, 1997), leaktight criterion (leakage rate of less than 10^{-7} standard cubic centimeters per second). Leaktightness is proven when the cask containment boundary is shown to meet ASME BPVC, Section III (American Society of Mechanical Engineers, 1998), Service Level A stress allowable during all normal transfer conditions.

The conditions to be evaluated for accident scenarios are pressurization of the cask and MCO and the performance of the cask and cask seal during postulated drop, puncture, and fire conditions. The impact accident is simulated by a free drop of 9.0 meters (30 feet) of the dry package or a free drop of 6.4 meters (21 feet) of the water-filled package onto a typical Hanford Site concrete surface. The puncture is a 1-meter free drop of the package onto a mild steel bar 15 centimeters (6 inches) in diameter. The thermal accident is simulated by exposure of the package to a 6-minute, 800°C (1475° Fahrenheit [F]) engulfing fire during transfer from the CVDF to the CSB, followed by a quench.

3.4.3 Identification of Hazard Controls

The CTS is used to transport the SNF about 0.5 mile from the K-Basins to the CVDF and then about 8 miles to the CSB. The CTS uses five MCO transfer casks and five dedicated semi-trailers. The MCO is positioned inside the cask before being loaded with SNF and remains in the cask until being removed for storage in the CSB. The loaded MCO transfer cask is moved on a dedicated semi-trailer attached to a standard tractor. The trailer provides the necessary supports and attachment points for securing the cask in the vertical orientation.

Each transfer cask consists of a forged stainless steel cylinder with nominal 7-inch-thick walls. Each cask is 170 inches long, with a 6-inch-thick welded stainless steel bottom head. The cask lid is stainless steel forging varying in thickness from 3.5 inches at the center to 3 inches at the edges. A containment boundary between the cask body and lid is formed by a butyl rubber O-ring face seal, located on the interface surface between the flange on the closure lid and the cask shell.

The MCO transfer cask is designed and fabricated to meet the leaktight criteria of ANSI N14.5 *Leakage Tests on Packages for Shipment* (American National Standards Institute, 1997), and maintain containment of the SNF through normal transfer and accident conditions at the Hanford Site. The cask closure lid uses the butyl rubber O-ring face seal to maintain leaktightness (10^{-7} standard cubic centimeters per second). There are three penetrations into the cask for two vent ports and a drain port. The larger vent port and the drain port are quick-disconnect couplings attached to coupling adapters, which are recessed from the exterior of the cask. These couplings are not considered to be containment boundaries. Leaktight containment for these ports is provided by cover plates and butyl rubber O-ring face seals.

The MCO transfer cask package must satisfy external release rate requirements for defined accident conditions. The structural analyses summarized in the SARP demonstrate that both the MCO transfer cask and the MCO itself satisfy ASME BPVC, Section III (American Society of Mechanical Engineers, 1998), Service Level D stress allowable during all accident conditions. Proper squeeze during accident conditions is also maintained on the cask lid seal by the cask lid flange bolts (squeeze is defined as the percent reduction in the cross-sectional diameter of the O-ring due to compression).

3.5 COLD VACUUM DRYING FACILITY

3.5.1 Definition of Functions and Requirements

The primary safety functions of the CVDF are to remove basin water from the MCO while maintaining the integrity of the containment boundary and preventing runaway thermal reactions due to rapid oxidation of the fuel.

As noted earlier, cold vacuum drying activities involve removing bulk water from the MCO, drying the fuel, backfilling the MCO with helium, and sealing it. When the MCO transfer cask arrives at the CVDF, the pressure within the transfer cask is measured to determine whether a significant pressure increase has occurred. In the event such an increase in pressure is detected, a likely cause is a rise in internal cask temperature resulting from rapid fuel oxidation. These conditions could indicate the early stages of a runaway thermal reaction (see Section 3.2.2.2).

If there is no evidence of such a pressure increase, the cask is vented. Because the MCO is vented to the cask interior, this venting also vents the MCO head space. After the cask and MCO head space are vented, the cask is backfilled with helium, and the cask shield cover is removed to permit processing of the MCO.

Processing of the MCO requires that a helium atmosphere be maintained inside the MCO and that its temperature be controlled. A vacuum system is needed to remove remaining free water from the MCO. This water from the K-Basins is handled as radioactive fluid that may contain fuel particles.

Potential releases of hydrogen and radioactive gases from the MCO need to be controlled and released in a safe manner. Because of the potential to release radioactive material, the CVDF ventilation system needs to provide positive flow from areas of lesser contamination into areas of higher contamination. On the basis of accident analyses and an issue report prepared by the Board's staff (Gwal, 1998), a safety-significant process exhaust system was identified, and backup power by a diesel generator has been provided (see Section 3.5.3.8).

3.5.2 Analysis of Hazards

Potential events occurring in the CVDF that could pose a challenge to off-site and on-site radiological evaluation guidelines were evaluated as design basis accidents. The following subsections address these potential accidents.

3.5.2.1 Gaseous Release

The bounding unmitigated scenario for this accident involves a pressurized release of helium gas and entrained contaminated particulates through a process line leak. The unmitigated consequences of this event do not exceed the off-site release limits, but do exceed the on-site risk evaluation guidelines. No safety-class features are required to mitigate this event. Safety-significant features of the design for coping with this event include portions of the process general supply/exhaust heat, ventilation, and air conditioning (HVAC) system and process bay local exhaust HVAC and process vent system, as well as differential pressure alarms for the process bays and process water tank room. Mitigated consequences of this event are well below both off-site release limits and on-site risk evaluation guidelines (see Table 3-1).

3.5.2.2 Liquid Release

The bounding unmitigated scenario for this accident involves a pressurized leak of water and entrained contaminated particulates from the process water conditioning piping. The unmitigated consequences of this event do not exceed the off-site release limits, but do exceed the on-site risk evaluation guidelines. As in the case of a gaseous release as discussed above, 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 differential pressure alarm for the process water tank room. Mitigated consequences of this event are well below off-site release limits and on-site risk evaluation guidelines.

3.5.2.3 MCO External Hydrogen Deflagration

The bounding unmitigated scenario for this accident involves 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 deflagration of the hydrogen gas. The unmitigated consequences of this event do not exceed the off-site release limits, but do exceed the on-site 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 system and process vent system (ductwork and HEPA filters) and special tools to limit the cask vent flow rate. Mitigated consequences of this event are well below both off-site release limits and on-site risk evaluation guidelines.

3.5.2.4 MCO Internal Hydrogen Deflagration

The bounding unmitigated scenario for this accident involves the ignition and deflagration of a hydrogen-air mixture inside an MCO. The unmitigated consequences of this event do not exceed the off-site release limits, but do exceed the on-site risk evaluation guidelines. No safety-class features are required to prevent or mitigate this event. Some safety-class features (i.e., multiple safety functions to detect process upsets, the safety-class helium system, portions of the tempered water [annulus] system) that prevent a runaway thermal reaction in the MCO (see Section 3.5.2.5 below), as well as certain overpressurization events, also help to prevent this accident. In this context, however, they play only a safety-significant role. Because the designated safety features prevent and mitigate this event, both off-site release limits and on-site risk evaluation guidelines are satisfied.

3.5.2.5 Runaway Thermal Reaction

The bounding scenario for this accident is initiated by loss of or diminished heat removal from an MCO. This condition could lead to an uncontrolled escalation of the chemical reaction within the MCO, resulting in excessive internal temperature. If unmitigated, the high temperatures of this scenario could lead to a continuous release of gas and contaminated particulates for an extended period of time. The unmitigated consequences of this event challenge the off-site release limits and exceed the on-site risk evaluation guidelines. Safety-class features selected to prevent this event include safety features to detect process upsets, the safety-class helium purge and isolation system, and portions of the tempered water (annulus) system. Mitigated consequences of this event are well below both off-site release limits and on-site risk evaluation guidelines.

3.5.2.6 MCO Overpressurization

The bounding unmitigated scenario for this accident involves 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 internal pressure in the MCO continues to increase until the MCO pressure boundary is breached or until the fuel or water is completely consumed. The overpressurization leads to a pressurized release of gas and contaminated particulates, 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 off-site release limits and on-site 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 psi(g) vent line, the 150 psi(g) 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 off-site release limits. Additional safety-significant features for confinement and filtration are identified to mitigate the on-site consequences to well below the on-site risk evaluation guidelines.

3.5.3 Identification of Hazard Controls

3.5.3.1 Monitoring and Control System

The CVDF FSAR describes the function and operation of the Monitoring and Control System (MCS) used by the operators for normal CVDF process control. This system is classified as general service. It provides indication, control, and alarms for various processing functions, including the following:

- Vacuum Purge System (VPS).
- Tempered water (annulus) system.
- Process Water Conditioning System (PWCS).
- General service helium system.
- Deionized water system.

The MCS, which interfaces with the above systems in each of the CVDF process bays, consists of interactive computer terminals, logic circuits, input/output modules, and panels, but does not include process sensors such as those for pressure, temperature, and flow. The system allows for simultaneous control via workstations connected by a local area network. The MCS monitors several systems, including radiation monitoring, stack monitoring, VPS chilled water, tempered water cooling, helium supply, and instrument air.

The system also controls MCO isolation valves under normal operations. These are the same valves used for isolation by the Safety-Class Instrumentation and Control (SCIC) System. However, the SCIC System will override control commands from the MCS. The MCS takes inputs from the SCIC System but is electrically isolated from it. The Board's staff has confirmed that the MCS controls normal process functions only, is electrically isolated from the SCIC System, and is designed to be overridden by the SCIC System's control signals.

3.5.3.2 Safety-Class Instrumentation and Control System

The CVDF FSAR identifies the SCIC System as a system whose failure could indirectly result in a condition adversely affecting the health and safety of collocated workers. The system also detects and controls functions designed to prevent challenges to off-site limits. Thus, the system is designated safety-class. It provides safety functions for four design basis accidents:

- MCO overpressurization.
- MCO runaway thermal reaction.
- MCO internal hydrogen deflagration.
- MCO external hydrogen deflagration.

The SCIC System performs three general functions to mitigate MCO upsets:

- MCO isolation and purge, which isolates the MCO and initiates safety-class helium pressurization and purge of the MCO.

- Removal of MCO excessive heat, which removes power to tempered water system heaters.

- Control room safety-class annunciation, which provides several safety-class and defense-in-depth alarms to operators in the control room.

The SCIC System is designed to monitor various system parameters and to place the MCO in a safe condition. As noted, it is electrically separate from the MCS, which is used for CVDF normal process control (see Section 3.5.3.1). The MCS takes inputs from and controls the same isolation valves as the SCIC System, but control signals from the SCIC System override those from the MCS. The SCIC System contains redundant channels, and is designed to operate under a single failure of one channel. The system contains circuitry for operational testing. Test switches for testing the functional performance of the system are provided. Calibration and additional functional tests are performed using a calibration and test computer not normally connected to the SCIC System. The system design specification for the SCIC System references the pertinent IEEE standards[3] for safety-class systems and complies with the requirements of DOE orders and implementation guides. The Board's staff has concluded that the SCIC System is designed with the required channel redundancy and separation.

In the enclosure (Gwal, 1998) to a letter from the Board dated December 1, 1998 (Conway, 1998), the staff addressed the issue of the small margin (0.9°C) between normal operating temperature and the set point for the high-level trip on cask annulus water temperature. The enclosure to the Board's letter also raised the issue that the existing alarm system for water level in the cask annulus may not be able to withstand a seismic event.

The calibration frequency for the high level trip on cask annulus water temperature has been increased from annual to quarterly, thus reducing the calculated switch errors to more acceptable values. In addition, the operating temperature has been reduced, providing a margin of 3.1°C between nominal values. The Board's staff has concluded that this margin is acceptable.

During September 2000, the Board's staff reviewed sensor error and set point calculations provided by project representatives for the set point and margins associated with water level alarms in the cask annulus. Based on that review, the staff has concluded that the set points are appropriate and that the robustness of the system has been adequately tested.

3.5.3.3 Tempered Water System

A portion of the tempered water (annulus) system contains safety-class piping and inti-siphon valves to retain a minimum water level above the elevation of the top of the spent fuel within the MCO. The tempered water system is used to heat the MCO to a temperature of 40 to 50°C prior to vacuum drying of the fuel. The system is connected to the annulus through the cask seal ring and a port near the bottom of the cask. The recirculating tempered water flows through the cask annulus from the bottom to the top for a "soak" period so the MCO contents can warm up, and the water temperature of the annulus can reach its operating range of 40 to 50°C. Following the soak period, the drain, purge, and drying sequence commences.

In the enclosure (Wille, 1998) to a letter to DOE dated March 18, 1998 (Conway, 1998), the Board's staff documented its concerns related to runaway reactions. Because the MCO is still filled with water as it is heated, the staff suggested that the process be revised to remove the water from the MCO before raising the initial temperature significantly above ambient conditions (nominally the temperature of the water in the basin), thus decreasing the potential for a runaway

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[3] These include IEEE Std 308-1991, *IEEE Standard Criteria for Class 1E Power Systems for Nuclear Generating Stations*; IEEE Std 323-1983, *IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations*; ANSI/IEEE Std 338-1987, *IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Safety Systems*; -344, *IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations*; IEEE Std 379-1994, *IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems*; IEEE Std 384, *IEEE Standard for Independence of Class 1E Equipment and Circuits*; IEEE Std 603-1980, *IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations*; and IEEE Std 627-1980, *IEEE Standard for Design Qualification of Safety Systems Equipment Used in Nuclear Power Generating Stations*.

thermal reaction. As an indication of the sensitivity of the reaction to temperature, heating the annulus water to a temperature of 75°C was acknowledged by project personnel and DOE-RL to be unsafe. The reason for heating the MCO to 50°C while still filled with water is to reduce the processing time for drying.

On August 31, 1998, a letter (Moniz, 1998) from DOE responded to the above issue. DOE's response stated that lowering the temperature at which the water is drained had been suggested during safety reviews as a means of providing additional margin against oxidation reactions, but the additional processing time required to heat a drained MCO had not been fully evaluated. DOE believed the drying process would be unacceptably slow because effective heating could not be provided after draining the water.

The Project FSAR, Annex B, *Cold Vacuum Drying Facility* (Fluor Hanford, 2000), addresses the controls that prevent temperatures from exceeding 50°C. The CVDF process requires that the normal drying operation and dryness testing for the MCO be performed at 40 to 50°C. If the inlet water temperature is too high, the safety-class temperature switches send a signal to the SCIC System, which deenergizes the tempered water (annulus) system heater. The SCIC System and associated safety-class instruments also detect process upsets and activate the safety-class helium system to provide an MCO purge and vent function.

The Interim SER for the CVDF, Annex B, raises the same issues regarding runaway thermal reactions. The items were closed by an analysis in Annex B of the FSAR, identifying several potential accidents that could lead to a runaway thermal reaction. The bounding accident, which is the design basis accident, is loss of water. The FSAR states that this accident requires simultaneous loss of helium flow, loss of cask-MCO annulus water, and continuous release through a failed process line. The FSAR estimates the frequency of this accident as beyond extremely unlikely, eliminating the need for additional mitigation features. The Board's staff believes no further analysis is necessary.

3.5.3.4 Process Water Conditioning System

Normal operation of the PWCS involves draining water from the MCO, processing the collected water through ion exchange modules (IXMs), and transferring the processed water to a 5,000-gallon storage tank. The PWCS is powered by two pumps, one running and one in standby. Radionuclides are removed using two self-shielding, single-use, disposable IXMs. After the process water has been sampled and the results verified, the water is transferred to the K-Basins via the Conditioned Water Shipping System.

The MCS controls all non-safety automated functions and alarming of the PWCS. The SCIC System has interlocks and trip control of the PWCS-to-MCO isolation valves. The PWCS isolation valves and associated piping are safety-class, the lines to the receiver tanks are safety-significant, and the remaining components are general service. The tanks in the IXMs are critically favorable by design.

3.5.3.5 Safety-Class Helium System

Five sources of helium are provided in each process bay of the CVDF. The general service helium system is supplied from a tube trailer, and the standby safety-class helium system is supplied from four helium gas cylinders. Helium is used in process operations to provide enhanced thermal conductivity, to purge the MCO of hydrogen and other gases, to pressurize the MCO to preclude further air ingress, and to provide an inert backfill of the MCO when drying is complete.

The essential safety function of the safety-class helium system is to inert the MCO and associated process piping upon actuation by the SCIC System during process upsets. The safety-class helium system prevents a runaway thermal reaction caused by insufficient heat removal from the inside of the MCO, in addition to preventing hydrogen deflagrations caused by the buildup of flammable concentrations of hydrogen and oxygen.

3.5.3.6 Vacuum Purge System

The VPS consists of isolation valves, piping, and instrumentation in conjunction with a residual gas analyzer, a condenser, a condenser water collection tank routed to the PWCS, a helium line to provide pressurization to remove MCO water, and a vacuum pump that connects to the process vent. After being drained of bulk water, the MCO is evacuated by the vacuum pump, and the condenser captures water vapor in the gas removed from the MCO. Upon indication that the most of the water has been removed, the leaktightness of the MCO is checked to verify that the MCO is ready for shipment.

The FSAR, Annex B, *Cold Vacuum Drying Facility* (Fluor Hanford, 2000), cites as safety-class components those that isolate the MCO from the VPS, monitor gauges, and provide an auxiliary vent path in situations in which the MCO pressure rise is greater than can be handled by the vent of the safety-class helium system. The safety-class vent path mitigates the effect of a pressurized release from the MCO due to a postulated overpressurization event. The pressure transmitters, pressure indicators, flow indication transmitters, and MCO rupture disk are safety-class components.

3.5.3.7 Confinement Systems

Portions of the ventilation systems listed below provide safety-significant functions to prevent or mitigate postulated accidents in the CVDF. Specific components or portions of the system providing the safety-significant function are discussed in the following paragraphs.

Process Bay Local Exhaust HVAC and Process Vent System. Two HVAC systems (process bay local exhaust HVAC and process vent system, and process general supply/exhaust HVAC system) provide confinement of airborne radioactive material within the radiologically controlled areas of the CVDF, as well as providing HEPA-filtered discharge via the CVDF stack. These systems maintain appropriate building ventilation zone pressures such that air flow is from less to more potentially contaminated areas. Each process bay also has an independent HVAC system for process bay recirculation that provides an outside air supply and HEPA-filtered recirculation for heating and air conditioning.

The primary confinement feature is the stainless steel MCO. The MCOs are transported to the CVDF inside the MCO transfer cask (see Sections 2.2.3 and 3.4.1). The mechanically sealed transfer cask forms a secondary confinement barrier during transportation and receipt. The primary confinement of airborne and liquid effluents from the SNF during processing in the CVDF is provided by the MCO and isolation piping.

The MCO is connected through three process lines to the VPS, general service helium system, deionized water system, and PWCS 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's contents from the secondary confinement provided by the building HVAC systems. 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 would be into the process bay or process water tank room. The system filters air from the process hoods, and vents air from the VPS and tempered water (annulus) system, as well as gases from cask venting and the safety-class helium system, prior to discharge from the facility.

The process vent portion of the process bay local exhaust HVAC and process vent system is classified as safety-significant and is designed to provide sufficient air flow to dilute the potential hydrogen releases to nonflammable concentrations. It provides secondary confinement at the top of the MCO, using an open-face process hood with a capture velocity of 125 feet per minute at the top of the MCO. The design of the process hood conforms to applicable requirements of DOE Order 6430.1A, *General Design Criteria* (U. S. Department of Energy) and the hood design recommended in the American Conference of Governmental Industrial Hygienists, Industrial Ventilation, 21st Edition, *A Manual of Recommended Practice* (American Conference of Governmental Industrial Hygienists, 1992).

Process General Supply/Exhaust HVAC System. During normal operation, secondary confinement is provided with a differential pressure established by the process general supply/exhaust HVAC system. A differential pressure is also maintained following a loss of normal electrical power through support of the standby power system when other ventilation systems are not operable. The safety-significant standby power system (see Section 3.5.3.8) supports the functions of the local exhaust system of maintaining adequate differential pressure for confinement in the process bays and reestablishing the minimum airflow rate required for hydrogen dilution in the local exhaust system ductwork during a loss of normal electric power.

Process Bay Recirculation HVAC System. 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 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.

Reference Air System. All exhaust systems are in operation during normal processing. Monitoring of the differential pressure by the safety-significant reference air system 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 safety-significant inlet damper of the process bay recirculation HVAC system is closed to increase flow through the doorway. The HVAC systems use isolation dampers to stop the flow of air in order to maintain confinement within the facility and preclude cross-contamination between areas during upset conditions.

3.5.3.8 Electrical Systems

Annex B of the CVDF FSAR states that electrical power is not required for any safety-class systems and that the only system requiring safety-significant power is the exhaust ventilation system. The local bay exhaust and process vent system is designed to provide confinement in CVDF process bays and dilution flow in the local exhaust duct upon loss of normal power. Safety-class SSCs fail safe upon a loss of power and do not require safety-related electrical power.

The normal electrical power distribution system provides power for the process and SCIC systems as well as other CVDF auxiliaries. A 13.8 kV on-site electrical system delivers power to a three-phase step-down transformer with Y-connected secondary at 480 V.

The standby power system consists of a 100 kilowatt (kW) diesel generator, an automatic transfer switch, a test load bank, and various support systems. It provides backup power to the process bay local exhaust HVAC and restart circuit, process bay heat trace, uninterruptible power supply system, and instrument air compressor. The standby power system is classified as safety-significant.

An October 21, 1998 staff issue report (Gwal, 1998), forwarded to DOE on December 1, 1998 (Conway, 1998), noted that although the ventilation system itself was designated as safety-significant, an exhaust fan in the system was classified as general service, and the system had no backup power. In February 1999, DOE responded by letter (Owendoff, 1999) that the exhaust fans would be classified as safety-significant and that backup power would be provided.

In a March 1999 letter (Conway, 1999) to DOE, the Board forwarded a staff issue report (Wille, 1999) indicating that the contractor had not fully accepted the safety-significant designation for all necessary equipment. The Board revisited this subject in a letter dated July 8, 1999 (Conway, 1999), emphasizing the need for safety-significant power for the CVDF.

In a September 1999 letter (Huntoon, 1999), DOE responded to the Board’s March 29 and July 8, 1999, letters (Conway, 1999) concerning the need for a standby power source. DOE’s response indicated that in May 1999, the project staff had approved a design change notice that added a diesel generator and transfer switch to the facility and provided a separate building for the diesel generator, located approximately 100 yards from the CVDF. The Board’s staff has confirmed the seismic qualification of the safety-significant standby power system in accordance with IEEE 344-1987, *IEEE Recommended Practice for Seismic Qualification of Class 1E*

Equipment for Nuclear Power Generating Stations (Institute of Electrical and Electronics Engineers, 1987).

During a June 2000 review of the safety-significant electrical power system, the Board's staff also raised an issue regarding the need for sequencing of loads on the diesel generator and the adequacy of the generator to start and support all the CVDF loads. During an August 22, 2000 video conference, project personnel presented load calculations, evaluation of transient time-current characteristics of the generator and major loads, and built-in time delays in the generator circuitry, to demonstrate that the capacity of the diesel-generator is not challenged during startup of the major loads. The Board's staff concurred with this analysis.

The same video conference included a discussion of a diesel trip on high cooling water temperature and the contractor's root-cause analysis of this event. Engineering evaluations disclosed that the conditions encountered were exacerbated by an undersized radiator on the diesel. In addition, the ambient heat load in the diesel-generator room was increased by presence of the load bank. A larger room exhaust fan motor was installed, the load bank was moved, and certain non-safety loads were removed from the generator bus.

Two tests were performed satisfactorily, verifying the capability of the diesel-generator to support the CVDF loads and successful starting of the local process bay exhaust fan. The generator was run for three hours with a 50 kW resistive load to demonstrate adequate performance under load. The second test verified that the diesel-generator would start within 10 seconds of loss of power and that the local exhaust fan would start and deliver 1,000 cubic feet per minute within 60 seconds.

During a 1998 site visit, the Board's staff had noted the lack of a lightning protection system at the CVDF's ventilation exhaust stack. DOE has since designed and installed a lightning protection system, as discussed in the FSAR. The staff has confirmed that the lightning protection system was designed in accordance with the National Fire Protection Association (NFPA) 780, *Standard for the Installation of Lightning Protection System*, 1997 Edition (National Fire Protection Association, 1997).

Comprehensive short-circuit, voltage profile, and coordination studies are essential to safeguard personnel and maintain a safe and reliable power system. These studies should be performed in accordance with appropriate IEEE standards.[4] Also, in accordance with the CVDF design requirements document, electrical system components are required to be coordinated for short-circuit capability, interrupting duty and capability, insulation levels, protective relaying, reliability, interchangeability, transformer and line voltage drop, stability under normal conditions, and restart upon power dips and outages. The Board's staff confirmed that these calculations and studies have been performed.

[4] IEEE-141, *IEEE Recommended Practice for Electrical Power Distribution for Industrial Plants*, and ANSI/IEEE 242-1986, *IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems*.

3.5.3.9 Fire Protection

The fire protection system for the CVDF includes the fire detection and alarm system and the suppression system. All fire protection system valves, components, instrumentation, and controls required to perform detection and suppression functions are designated general service and are designed and qualified for seismic Performance Category (PC)-1. The facility is automatically monitored for fire and smoke by ionization-type smoke detectors. The fire alarm system is designed and installed in accordance with NFPA-72, *National Fire Alarm Code,* 1996 Edition (National Fire Protection Association, 1996). Fire alarm signals are transmitted to the Hanford Fire Department via a radio fire alarm reporter. Manual pull stations are also provided at emergency exits. Automatic sprinkler protection has been provided throughout the facility in accordance with NFPA-13, *Installation of Sprinkler Systems,* 1994 Edition (National Fire Protection Association, 1994). The water supply system is adequate to meet the design flow and pressure demand of the sprinkler system. In addition, fire hydrants are located just outside the CVDF at two corners.

The FHA pertaining to the CVDF meets the content requirements of DOE Order 5480.7A, *Fire Protection* (U.S. Department of Energy, 1993). The FHA contains recommendations to resolve identified deficiencies before processing in the CVDF begins. Resolution of these deficiencies may be completed under the equivalency/exemption process.

The Board's staff has concluded that the fire protection system for the CVDF is adequate, based on the FHA and review of the system design.

3.5.3.10 Structural and Other Engineering Aspects of Hazard Controls

The CVDF is a temporary facility with a design life of 5 years (see Section 2.2.4). The facility is characterized as Hazard Category 2; the building features are categorized as safety-significant.

The CVDF process bays are designed to preclude damage or functional impairment of safety-class systems and components within the bays under all conditions. The process bays also provide tertiary confinement in conjunction with the HVAC systems during facility operations. On the basis of these functions and requirements, the process bays are designed to meet the requirements of PC-3 for protection against natural phenomena hazard (NPH) design loads imposed by potential seismic, straight wind, tornado, volcanic ash, flooding, lightning, and snow events. PC-3 seismic design spectra are generally acceptable for a Hazard Category 2 facility such as the CVDF (see Figure 3-1 for the CVDF design response spectra). The process support area and the process water tank room are designed to meet NPH PC-2 requirements, and the administration building PC-1. PC-1 and -2 SSCs are designed to meet the requirements of the 1994 Uniform Building Code™, Volume 1, *Administrative, Fire- and Life-Safety and Field Inspection Provisions* (International Conference of Building Officials, 1994). The Board's staff has identified no issues associated with the NPH design criteria for the CVDF.

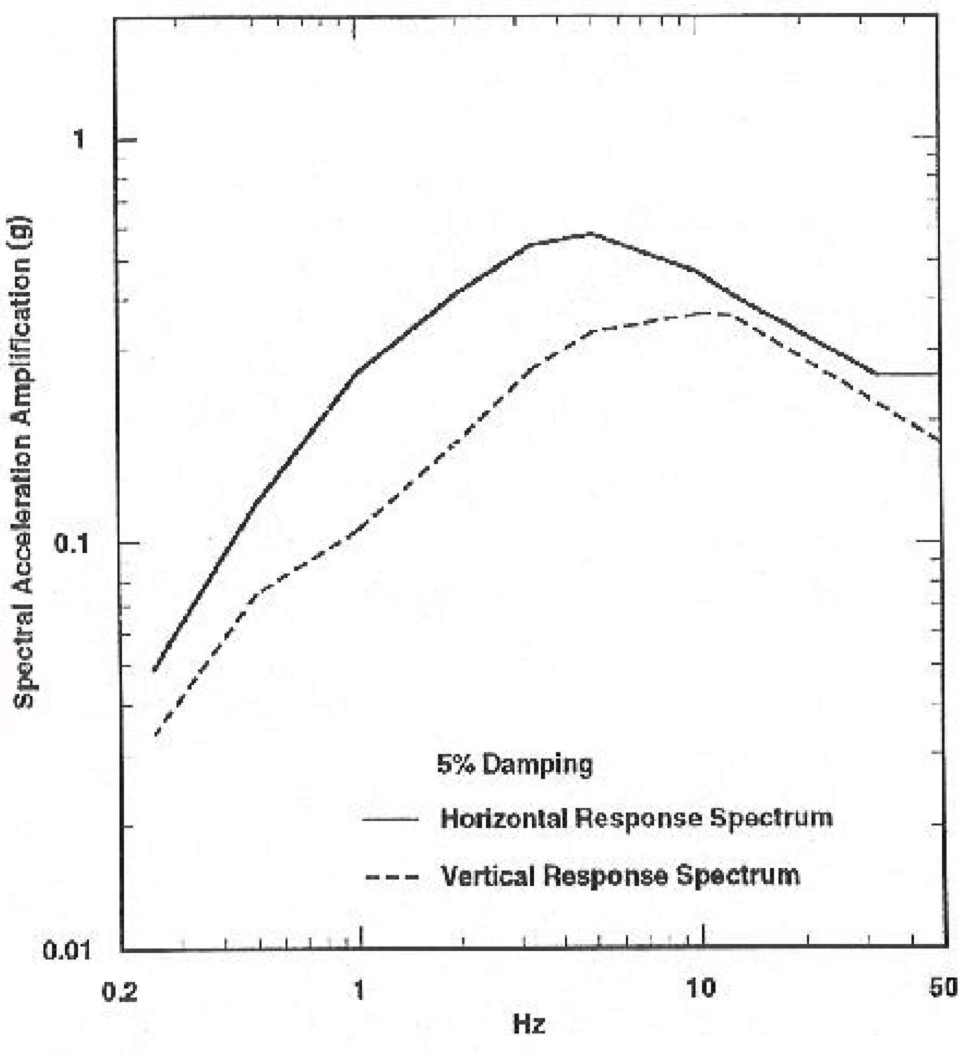

Figure 3-1. Performance Category 3 Response Spectra for the Cold Vacuum Drying Facility

Source: Fluor Hanford Inc., HNF-3553, Annex B, Revision 1 (Fluor Hanford, Inc., 2000)

3.6 CANISTER STORAGE BUILDING

3.6.1 Definition of Functions and Requirements

The primary safety functions of the CSB are to receive and transfer MCOs within the building; store them in below-grade storage tubes; provide a shielded sampling/weld station; and maintain adequate shielding, passive cooling, and containment of the MCOs during interim storage.

The CSB provides for the receipt, sampling, monitoring, and interim (40 years) dry storage of SNF contained in MCOs. The facility comprises an above-ground steel-frame building that encloses the operating and loading/loadout areas, and three equal-sized below-ground reinforced concrete vaults. The below-ground walls are sized for radiation shielding. The basemat is nominally 5 feet, 6 inches thick; the below-ground exterior walls are 4 feet, 6 inches thick; and the walls separating the three vaults are 3 feet thick. The concrete vaults are covered at grade level by a 5-foot-thick concrete operating deck. The operating deck contains full-thickness embedded steel sleeves that receive the tube plugs and connect to the storage tubes (about 40 feet long and 28 inches in diameter) from below. Only Vault 1 is equipped with steel tubes for storage of the MCOs. The MCOs are the first barriers providing interim containment and confinement of the enclosed SNF. Vault 1 is cooled by natural convection through an above-grade inlet structure, through a below-grade concrete intake plenum, to an exhaust plenum and exhaust stack (see Figures 2-7 through 2-9).

The MHM is used to move the MCO from the receiving area to a storage tube location. After the MHM is indexed over the proper storage tube, it removes the storage tube plug and lowers the MCO into the storage tube. Impact absorbers are placed in the bottom of the storage tube and between MCOs (two MCOs can be placed in a single storage tube) to mitigate the effects of a potential cask drop. The MHM is also used to move MCOs to and from the sampling/weld station, as required (see Figures 2-10 and 2-11).

The partially constructed foundation of the canceled Hanford Waste Vitrification Plant (HWVP) was incorporated into the design of the CSB. This decision to use the earlier work on HWVP had a significant effect on the seismic design criteria for the CSB (see Section 3.6.3.1).

3.6.2 Analysis of Hazards

The CSB is appropriately categorized as Hazard Category 2, and designed to NPH PC-3 requirements. NPH design loads considered are seismic, straight wind, tornado, volcanic ash, flooding, lightning, and snow. In addition to NPH, the design addresses hazards from aircraft impact, transportation accidents, and the effects of operations at nearby facilities. The safety-class structural features of the CSB are discussed in Section 3.6.3.

A beyond-design-basis event scenario assumes that the passive cooling air flow through the CSB vault is substantially reduced. If the air flow were reduced enough for a sufficient time period, the MCO walls or the concrete of the CSB vault could exceed their respective design temperature limits. Violation of design temperature limits is not considered credible because of passive design features associated with the vault, vault intake structure, and vault exhaust stack, coupled with the extensive amount of time available to alleviate blocked vault air intake or exhaust pathways. Passive cooling of the MCOs is a safety-class feature.

In its assessment of postulated accidents for the CSB, DOE-RL concluded that (1) the SAR analysis is bounding; (2) the postulated events are characterized in a conservative manner; and (3) the conservatism more than compensates for the consequences of any residual uncertainty caused by process variability at the K-Basins and the CVDF, as well as the imperfectly understood physical and chemical behavior of the SNF during interim storage in MCOs.

DOE-RL found the contractor's development of scenarios and assumptions that could potentially result in the occurrence of a criticality event in the CSB to be very conservative. In addition, restrictions on water sources in the CSB further limit the possibility of inadvertent criticality. DOE-RL concluded that a criticality event in the CSB is not credible and that its frequency of occurrence is much less than 1.0×10^{-6} per year. (For a generic discussion of potential accidents involving inadvertent criticality, see Section 3.2.2.1.)

Safety-significant design features and controls for safe handling of MCOs and for worker safety in the CSB are provided. These features, which also provide defense in depth, include radiation shielding, design of the operations area shelter and other equipment to prevent damage to safety-class features during design-basis events, and means to maintain confinement during venting of the transportation cask and obtaining gas samples in the sampling/weld station.

3.6.3 Identification of Hazard Controls

3.6.3.1 Structural Features

The safety-class features relied upon in the facility safety basis to protect the MCO are as follows:

- Subsurface structures, including vaults, air intake, and exhaust plenum.
- Storage tubes, base slab embeds, and impact absorbers.
- At-grade structures, including operating deck.
- Intake structure and exhaust stack.
- MHM seismic restraint system, including MHM rails.

Two early decisions regarding the CSB had substantial effects on the project. First, the fast track adopted for the project dictated that development of design criteria, design of SSCs, and construction proceed on an essentially concurrent schedule. This resulted in numerous design compromises and in-process changes during construction, with resultant impacts on the technical, cost, and schedule baselines for the project. Second, DOE's adoption of a policy for achieving nuclear safety equivalent to that provided by NRC requirements, in conjunction with DOE's own requirements, initially led to confusion, particularly with regard to the design requirements for protection against NPH set forth in 10 CFR Part 100, *Reactor Site Criteria* (National Archives and Records Administration, 1999). These two issues were brought to DOE's attention in the enclosures (Hadjian 1995, 1996) to two letters from the Board dated December 15, 1995 (Conway, 1995), and June 11, 1996 (Conway, 1996), respectively.

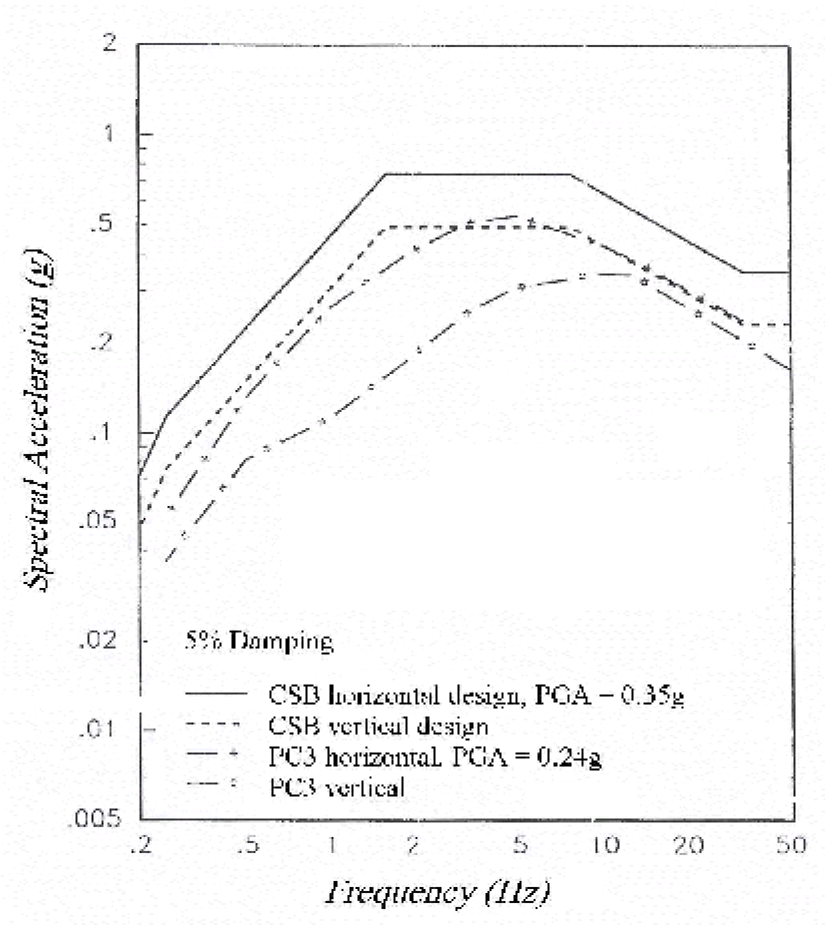

Figure 3-2. Canister Storage Building, Newmark and Hall Median Response Spectra at 0.35 g Horizontal and 0.23 g Vertical, Compared with Performance Category 3 Spectra, 5 Percent Damping

Source: Westinghouse Hanford Company, WHC-SD-SNF-DB-009 Revision 4, *Canister Storage Building Natural Phenomena Hazards* (Westinghouse Hanford Company, 1996).

After considerable consultation between the Board's staff and representatives of DOE and the SNFP, the issue involving the impact of the fast track on project baselines was resolved. To ensure that the fast tracking would not adversely affect the safe performance of the CSB, the Board's staff suggested that a confirmatory analysis be conducted in a timely fashion, before construction had proceeded too far for appropriate corrective actions to be taken. DOE agreed to perform such an analysis (Alm, 1996).

The fast-track schedule resulted in several analysis deficiencies that required attention in mid-1996 while construction was in progress. The following are examples of these deficiencies:

- The total structure, including embedment in the ground effects, was not modeled
- Some significant loads were not considered (e.g., tornado/wind loads and missiles)
- Outdated standards were used
- Unproven analysis methods were used for seismic excitation of embedded structures
- Three fully loaded vaults and three sets of ventilation stacks were assumed in the analysis when only one would be built
- Thermal loading of all three vaults simultaneously were assumed in analysis when only Vault 1 would be thermally loaded

After several meetings involving DOE, the contractor, and the Board's staff, these analysis issues were properly identified and corrected (Alm, 1996). Examples of these analysis issues include:

- The seismic excitation of the significantly embedded substructure was initially analyzed using a new, unverified computer code that is not able to account for soil-structure interaction effects directly. A more appropriate code was subsequently used for analysis of soil-structure interaction. The Board's staff performed its own analysis of how the seismic input motion is transmitted to the relatively rigid below-ground structure. As a result of discussions between the staff and SNFP personnel, the contractor made adjustments to the final in-structure response spectra. The in-structure response spectra are used for the design and qualification of equipment and components.
- Analysis of the CSB with thermal loads in all three vaults would not correctly reflect the worst design conditions with only one vault containing SNF, as will be the case for the foreseeable future. Moreover, given the uncertainty of future CSB loading (one, two, or three of the vaults containing SNF), a checkered loading pattern was required, and used, to capture controlling design conditions throughout the structure.
- Other deficiencies identified by the Board's staff have been rectified. For example, as a result of the staff's review of the details of the deck design, the original arrangement of the sleeves was modified to enhance the capacity and ductility of the deck structure. The staff also performed significant reviews of the MCO drop on the tube-sleeve assembly to ensure that confinement would not be breached following an accidental MCO drop. Design modifications were introduced accordingly.

Because deficiencies identified by the Board's staff have been rectified, including a confirmatory structural analysis, and corrective actions taken, the staff concludes that, as presented in the FSAR, the CSB is adequately designed and constructed to perform its structural functions safely.

The issue of nuclear safety equivalency with NRC requirements was not fully achieved. When NRC requirements were not considered to be cost-effective, alternative design criteria were selected, consistent with the DOE policy. For example, even though seismic loads are the only significant loads considered in the CSB design, an exception was taken to the requirements of Appendix A to Part 100, *Seismic and Geologic Siting Criteria for Nuclear Power Plants*, 10 CFR (National Archives and Records Administration, 1999). Similarly, an exception was taken to the NRC requirements for protection against tornado missiles, based on probabilistic arguments. Although the NRC nuclear safety equivalency achieved was incomplete, the extent to which it was achieved was identified and justified by the project. Despite the anomalies, the Board considers that DOE Orders and standards provide an adequate basis for design of this facility.

The safety-class SSCs are designed using the seismic design spectrum specified in NUREG/CR-0098, *Development of Criteria for Seismic Review of Selected Nuclear Power Plants* (U.S. Nuclear Regulatory Commission, 1978), with the peak ground acceleration fixed at 0.35 times the acceleration due to the force of gravity. The design is considered sufficiently conservative since, in general, this spectrum is about 1.5 times more intense than the latest PC-3 site spectrum, and a PC-3 seismic design spectrum is generally acceptable for a Hazard Category 2 facility such as the CSB. The selected seismic design spectrum was the design requirement for the canceled HWVP. Since part of the below-ground structure was already constructed, it was deemed prudent to complete the design using the original HWVP design spectrum. A comparison of these spectra is shown in Figure 3-2.

During early design stages for the CSB, two projected life spans for the facility, 40 and 75 years, were being used simultaneously for different purposes. If the CSB could potentially be used for 75 years for interim storage, probabilistically based design loads must be upgraded to achieve the stated reliabilities. The Board's staff did not consider the approach of revisiting this issue after 40 years to be appropriate, since with this logic, one could decide to revisit any design annually, and thus justify the use of smaller loads for the initial design. After several discussions, DOE decided on a design life for the facility of 40 years (Alm to Conway, July 15, 1996).

3.6.3.2 Passive Cooling

The MCOs containing the SNF are stored inside storage tubes arrayed in Vault 1, using naturally circulating air to remove decay heat from the fuel elements. The storage vault is passively cooled by convective air flow around the outside of the storage tubes. The height difference between the operating vault floor and the air exhaust stack (169 feet above the operating vault floor) provides a strong chimney effect. The top of the inlet plenum opening is 7 feet below the top of the exit plenum, providing additional motive force for heated air to expand preferentially 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; out of the vault into the outlet plenum; and thence up the exhaust stack, where the temperature is monitored. It then discharges into the atmosphere.

The computational fluid dynamics model used in the SAR shows that the temperature difference between the highest vault air temperature (hot spot) and the outlet air temperature is only 9°F. The design of the cooling system limits the external temperature of the MCO to a maximum 270°F wall temperature to ensure maintenance of the fuel centerline temperature limit of 1,112°F and the concrete temperature limit of 150°F. Air inlet and exhaust temperatures and the flow rate are monitored using the distributed control system.

The thermal analysis for the CSB vault presented in the SAR addresses natural air convection flow due to the heat generated by nuclear decay of the stored SNF. The analysis shows that intensive air movement takes place in the vault zone where the tubes are located, and that the fuel centerline temperature limit of 1,112°F and the concrete temperature limit of 150°F are not exceeded. When the Board's staff questioned the validation of the computational design through physical measurements and tests, the contractor cited data available from experience in France, where a similar facility was built. The CASCAD facility in Caderache, France, in which SNF can be stored for up to 50 years, provides dry storage of fuel elements in stainless steel canisters that are stored in wells (tubes) below ground level with passive natural convection cooling. This facility is very similar in design to the CSB. The final design of the CASCAD facility, which started up in 1990, has been proven by actual field measurements.

During the design phase of the CASCAD facility and prior to the start of construction, a series of thermal tests was conducted to confirm the thermal analysis and validate the vault cooling design. Analyses covering heat transfer around a storage tube and the overall heat dissipation process were validated using a full-scale model of a complete tube (at 1000 watts) and a one-twelfth scale model (mockup) of the vault and cooling system, respectively. The results of the validation suggest the benefit of combining flow and thermal models of the storage facility with a detailed thermal model of a tube to perform a complete analysis. There was reasonably close agreement between the test and simulation results.

On the basis of the above considerations, the Board's staff believes the analysis for the CSB has adequately taken into account those variables which ensure that the fuel centerline temperature limit of 1,112°F and the concrete temperature limit of 150°F will not be exceeded. The French experience is so similar that additional testing is not warranted.

3.6.3.3 Building Ventilation

The HVAC system, which is part of the CSB confinement system, is classified as general service. It provides a controlled pressure gradient flow of air from uncontaminated areas to potentially contaminated areas of the building and out through HEPA filters and a monitored exhaust. During cold weather conditions, electric heating elements maintain the air temperature in the operating area within the normal range. For warm weather conditions, external, air-cooled condenser compressor units perform a comparable function.

3.6.3.4 Multi-canister Overpack Handling Machine

The MHM, which comprises a bridge and trolley system and a shielded cask and turret system, weighs approximately 990,000 pounds and is 17 feet high from the operating deck to the top of the trolley rails (see Figures 2-10 and 2-11). The machine is designed to transport an MCO safely within the service and operating areas of the CSB. It is also used to remove and replace storage tube plugs, the service station shield hatch plate, the sampling station shield plug, and impact absorbers.

Operation of the MHM is governed by a system of electrical interlocks designed to prevent or preclude operational errors that could result in radiological consequences or facility damage. The MHM is designed to prevent lateral displacement that would shear an MCO as it is being raised or lowered inside the throughport. A system of drive control interlocks prevents inadvertent activation of both bridge drive motors and the trolley drive motors unless the interlock sensors report that safe operation can be conducted. This interlock-permissive system greatly reduces the chance of damage to an MCO.

An August 22, 2000, video conference included a discussion of the existing safety interlocks for the MHM and their conformance to applicable codes and standards. The original design for the MHM called for redundant, interlocked safety-class controls. In an earlier review, the Board's staff had noted deviations from the provisions of IEEE Standard 384-1992, *IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits* (Institute of Electrical and Electronics Engineers, 1992). Subsequently, the contractor completed a new hazard analysis for the MHM and downgraded the classification of the interlocked controls to safety-significant. On the basis of the revised hazard analysis and the downgraded safety classification of the interlocks, the Board's staff concludes that there are no remaining open issues associated with MHM controls.

3.6.3.5 Weld Station Design

The functions of the weld station are to (1) sample the MCO for pressure and gas composition, and refill with helium if necessary; (2) weld the cover assembly on the MCO; (3) nondestructively inspect the weld; and (4) leak test the weld. The weld station is designed so that most of the MCO is in a pit below floor level with its top extending about 4 feet above the floor. Extra shielding is provided around the exposed regions of the MCO to protect the welders. An automatic gas tungsten arc welding process is used to weld the cover assembly to the MCO. The weld joint is designed for ultrasonic inspection, if necessary, but volumetric liquid penetrant inspection of the root, intermediate, and final layers is planned to be used as the weld acceptance inspection procedure. Volumetric liquid penetrant inspection is acceptable under the ASME BPVC. The cover assembly has a port for supplying backing gas during welding and for use during the final leak check of the cover assembly weld. A cover plate is welded over this port when leak testing is complete. The MCO is then placed in interim storage.

3.6.3.6 Electrical and Control Systems

Electrical and control systems described in the CSB FSAR include the normal electrical power system, an uninterruptible power supply (UPS), and a distributed control system (DCS). These three systems, which are classified as general service, are discussed below.

Normal electrical power for the CSB is supplied from an on-site 13.8 kV power source, reduced to 480 V by a step-down transformer with a Y-connected secondary, and distributed to four motor control centers. A UPS that can supply up to 20 kVA to instrumentation panels and the DCS is integrated into the normal electrical system. These panels power a number of health physics monitoring loads, telephones, and the fire alarm system. The UPS is also designed to pick up these loads upon a loss of normal power without overloading.

Portable diesel generators are available for additional backup power. On the basis of an analysis showing that no safety-class or safety-significant SSCs require backup power (discussed in Section 3 of the CSB FSAR), the contractor has determined that a permanently installed diesel generator originally designed for the CSB is no longer needed. That diesel generator has now been moved to the CVDF, in its own building.

In the enclosure to a letter dated March 29, 1999 from the Board (Conway, 1999), the staff noted that the original FSAR stated that two lightning rod assemblies are provided at the top of the exhaust stack, yet none were actually installed. The revised FSAR cites provisions of NFPA 780, *Standard for the Installation of Lightning Protection Systems*, 1997 Edition (National Fire Protection Association, 1997), which states that stacks with a metal thickness of more than 3/16 inch do not require air terminals or down conductors. On the basis of the current revision of the FSAR, which states that no stack has a metal thickness of less than 3/8 inch, the staff confirmed that the CSB lightning protection system is in accordance with NFPA 780, *Standard for the Installation of Lightning Protection Systems*, 1997 Edition (National Fire Protection Association, 1997).

Comprehensive short-circuit, voltage profile, and coordination studies essential to safeguard personnel and maintain a safe and reliable power system have been performed in accordance with applicable IEEE standards.[5] Electrical system components have been coordinated for short-circuit capability, interrupting duty and capability, insulation levels, protective relaying, reliability, interchangeability, transformer and line voltage drop, stability under normal conditions, and restart upon power dips and outages, in accordance with the CSB design requirements document.

The control systems at the CSB include the DCS and several safety-related control systems and interlocks. The main function of the DCS is focused on facility monitoring. The system displays a number of parameters, including area radiation, ventilation flow, alarm conditions, and several others. The DCS performs only one control function (operation of the helium dilution control valve), and provides archived storage and trend analysis for monitored

[5] These include IEEE-141, *IEEE Recommended Practice for Electrical Power Distribution for Industrial Plants*, and ANSI/IEEE Std 242-1986, *IEEE Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems*.

parameters. The CSB System Design Description states that the DCS has "no direct support role" regarding safety-significant or safety-class systems.

Several control functions and interlocks have been identified in the CSB SAR as safety-related, including the following:

- Seismic detection and MHM power disconnect system (safety-class).
- MHM collision avoidance system (safety-significant).
- Receiving crane positioning and interlock control system (safety-significant).
- MCO hoist and grapple (safety-significant).
- A number of safety-significant MHM interlocks.

The CSB FSAR lists appropriate IEEE and NFPA standards for these systems. The Board's staff has confirmed that the design of the safety-related interlocks provides for the required separation and isolation.

3.6.3.7 Fire Protection System

The CSB is provided with automatic sprinkler protection in the operations support area, which is in a building adjacent to the operations area proper. By design, there is no sprinkler protection in the operations area,[6] a provision approved by DOE-RL in an exemption granted in February 2000. Therefore, any fires in the container handling system must be fought manually or permitted to burn out. The CSB fire protection system is classified as general service. The building is of noncombustible construction, as required by the applicable DOE Orders. Additional fire protection features include fire extinguishers and a rated fire barrier between the operations and operations support areas. Combustible controls are provided by implementation of the FHA requirements through the fire protection program. No fire protection TSRs are provided. The facility has cleared vegetation from a 60-foot-wide space around the building to defend against potential range fires.

The fire alarm features include a very early smoke detection alarm system; 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 upon receipt of any alarm. The fire alarm system can also be activated using a manual box. The water supply system meets the requirements of DOE Order 6430.1A, *General Design Criteria* (U.S. Department of Energy, 1989). A freeze protection system is provided for the sprinkler system. A cathodic system for the underground lines is installed to prevent corrosion of the carbon steel water lines. The FHA for the CSB meets the content requirements of DOE Order 5480.7A, *Fire Protection* (U.S. Department of Energy, 1993).

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[6] This was done to prevent flooding of the MCO resulting from inadvertent activation of the sprinkler system.

The Board's staff has concluded that the fire protection system for the CSB is adequate, based on the FHA and review of the system design.

4. PROGRAMS FOR ENSURING THAT HAZARD CONTROLS ARE PROPERLY IMPLEMENTED

The work involved in the basic Integrated Safety Management function of Performing Work, as it is applied to the design and construction phase, encompasses those activities involved in design, procurement, fabrication, installation, and construction of safety-related SSCs and the programs that direct and control those activities (e.g., safety analysis, project management, quality assurance, and configuration management programs). These programs control the design effort as the project progresses from conceptual through final design. They also control acquisition, fabrication, installation, construction, and testing activities as the project becomes a physical reality.

4.1 SAFETY ANALYSIS REPORT PROGRAM

The SNFP started with an aggressive schedule for design and construction, even incorporating a partially constructed foundation from a previous project for CSB (see Section 3.6.3.1). To accommodate this approach, a phased safety analysis development was implemented. It was therefore necessary to obtain DOE's acceptance of safety analyses in a progressive fashion as the designs moved forward and approvals were needed for procurement, increments of construction, and installation. The approach required considerable effort on the part of DOE and the contractors to identify the appropriate information for inclusion at each stage of design development. The lack of a typical Preliminary Safety Analysis Report for a facility made it difficult to perform an integrated review of the facility, and resulted in a number of conservative design assumptions. Enabling assumptions were identified in the safety documentation, and considerable project effort was later required to identify and confirm that these early assumptions were valid.

The FSARs prepared by the project were developed in accordance with DOE Order 5480.23, *Safety Analysis Reports* (U.S. Department of Energy, 1992), and DOE-STD-3009-94, *Preparation Guide for U. S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Report*, (U.S. Department of Energy, 1994). The FSARs for the various facilities were completed and approved by DOE in early 2000, although many design changes were still being incorporated as final safety reviews were in progress. Although the use of a phased safety analysis approach proved difficult to implement effectively in practice, the FSARs ultimately produced by the project were of high quality and appropriate for use in an authorization basis.

4.2 PROJECT MANAGEMENT

The difficulties experienced by the project by 1997 were due largely to inadequate project management, as identified in DNFSB/TECH-17 (Defense Nuclear Facilities Safety Board, 1997). A significant reorganization of the project occurred in June 1998 when an integrated contractor project team was established, to which subcontractor personnel were directly assigned. A revised project baseline cost and schedule for the project was issued by the contractor in June 1998 and was approved by DOE-RL in December 1998. Since that time, the project has been managing to this baseline with regard to allocation of funding and schedule contingencies. This marked improvement in project controls is a direct result of having a realistic and justifiable baseline, with appropriate management attention to resolution of emerging technical issues.

4.3 QUALITY ASSURANCE

Quality assurance activities for the SNFP are required to comply with 10 Code of Federal Regulations (CFR) 830.120, *Quality Assurance Requirements* (National Archives and Records Administration, 1999), and DOE Order 5700.6C, *Quality Assurance* (U. S. Department of Energy, 1991). The SNFP has experienced difficulties in carrying out quality assurance activities, usually as a result of inadequate execution of work by design, fabrication, and construction personnel. This was noted in the 1999 Annual Report, *Price Anderson Nuclear Safety Enforcement Program* (U.S. Department of Energy), February 2000, which reported imposition of a civil penalty against the contractor for failure to meet the requirements of 10 CFR 830.120, *Quality Assurance Requirements* (National Archives and Records Administration, 2000). As an example of continuing quality assurance difficulties, cleanliness problems were discovered during 1998 in the first process skid during proof-of-process testing in the CVDF. The procurement specification for the test article did not specify cleanliness requirements for all of the process components, and grit and metal particles were found in the installed system. A review of the procurement specifications for all project components was then undertaken to ensure that appropriate cleanliness requirements were included. In August 2000, however, the safety-class helium system in the CVDF had difficulty holding a vacuum, a problem that was traced to scoring of the valve sealing surfaces of the isolation valves. A project review revealed that the cleanliness requirements specified were inadequate for vacuum and helium service. A flush of the vacuum purge system was required to restore proper cleanliness.

Deficiencies in welding quality assurance in the DOE complex, including the SNFP at Hanford, was the subject of a Board letter (Conway, 1999). In response, DOE is planning a complex-wide review of the effectiveness of site quality assurance programs, including those in place at Hanford. Results from this review and the quality assurance experience from K-West Basin activities should be applicable to the remaining K-East Basin efforts.

4.4 CONFIGURATION MANAGEMENT

Configuration management is an integral part of the design and construction phase of a project. It is performed to demonstrate that the as-built facility meets the specified design requirements and that the project documentation provides evidence for this conclusion. The SNFP experienced difficulties in achieving appropriate configuration management during the design and construction phase. As an example, during a delayed design review of the CVDF while the facility was under construction, the project could not demonstrate compliance with all of the specified requirements, such as DOE Order 6430.1A, *General Design Criteria* (U.S.

Department of Energy, 1998). Subsequently the project established a compliance matrix program for all of the SNFP facilities to document and demonstrate the compliance of all SSCs, as designed and constructed, with applicable requirements in DOE Orders and referenced standards.

4.5 EMERGENCY MANAGEMENT

Provisions for responding to potential emergencies, described in Chapter 15 of the FSAR, are straightforward. Emergency preparedness and planning measures for each of the SNFP facilities (K-Basins, CVDF, and CSB) and the on-site transportation of MCOs are compatible with the Hanford Site Emergency Plan and in compliance with DOE Order 151.1, *Comprehensive Emergency Management System* (U.S. Department of Energy, 1995) and the related guides.

The Board's staff has observed the performance of the Hanford Emergency Response Organization (ERO) on numerous occasions, including Exercise Oz, which involved a scenario in the K-Basins (Deplitch, 1995). Although none of these exercises were error-free, the ERO has, in general, been effective in dealing with simulated emergencies. Observed weaknesses have been associated with execution, rather than deficiencies in preparedness or in advance planning for coping with potential emergencies.

On the basis of its review of the FSAR and the Safety Evaluation Report, as well as its observation of emergency exercises and drills at the Hanford Site, the Board's staff has concluded that adequate preparations have been made for coping with potential emergencies involving any of the facilities and activities in SNFP.

4.6 TESTING

4.6.1 Supporting Tests

A sampling and test program for characterization of sludge was necessary for design and analysis of systems and components. K-W Basin sludge has two sources: (1) canister sludge and (2) floor sludge. Since the K-W Basin is lined and contains sealed canisters, floor sludge consists almost entirely of dirt and debris resulting from atmospheric settling of dirt. This sludge stream is of little safety concern. However, canister sludge, once believed to be minimal as a result of the use of potassium nitrite as a corrosion inhibitor during the canning process, does present both safety and operational issues.

With the discovery that sludge was present in closed canisters stored in the K-W Basin, the IWTS design required a much more detailed consideration of sludge properties. This issue was raised in DNFSB/TECH-17 (Defense Nuclear Facilities Safety Board, 1997) since, at that time, very little was understood about the physical and chemical properties of the sludge. This lack of understanding brought the feasibility of the design concept into considerable question. This situation resulted from initial sludge characterization studies that were poorly designed and qualitative in nature, and failed to provide the information needed to eliminate the risks associated with the performance and safety of the system. For example, understanding of the amount of metallic uranium in the sludge would have provided a more complete picture of the criticality risk and made it possible to identify the likelihood of runaway uranium oxidation reactions. As a result of not having this information, considerable effort was expended to evaluate these hazards, as well as to provide information related to system operability issues (e.g., particle size distributions to be used in design validation).

An attempt to better characterize canister sludge was completed in early 1998. By then, however, the IWTS design had been completed, and the system was being fabricated. Furthermore, these studies sampled only nine canisters, recovered extremely small volumes (from 2.5 to 27.0 milliliters), and provided information of little value for safety analysis and design verification. Nevertheless, results from these samples are instructive since the analysis did attempt to determine the particle size distribution. In contrast with the sludge in the K-East Basin, most of the particles in the canister sludge in the K-W Basin are less than 1 micron in diameter. Thus, the IWTS may experience decreased efficiency in separating these small particles from the water, and the basin clarity could adversely affect operator efficiency and increase dose rates to the workers.

4.6.2 Preoperational Tests

A timely and effective test program to support design and safety analysis is important to project success. The preoperational testing program for the SNFP was formally organized and directed by the operations organization. It was managed by the startup manager, with supporting roles played by staff from the operations, engineering, and construction organizations. This Joint Test Group provided technical review and approval of all test plans and results. Each facility (K-Basins, CVDF, and CSB) had a dedicated test director tasked with ensuring that testing was conducted within procedural requirements and that all testing objectives were met. In addition, each system had a dedicated design authority assigned by the engineering organization to ensure that test specifications were correct and test objectives were met. Each test was developed and conducted by a startup engineer working in conjunction with the system design authority. The overall testing program was well conceived, but somewhat complicated. This resulted in some misunderstanding regarding individual responsibilities when rapid recovery actions were required (e.g., recovery actions following cessation of testing due to failed components or the resolution of test deficiency/noncompliance reports). As an example, the IWTS booster pump was failing continually in the automatic mode of operation. The project elected to operate the IWTS system in manual mode to support continued testing, including the start of the Readiness Assessment for hot operations during the Phased Startup Initiative.

The aggressive schedule resulted in less than adequate rigor in certain aspects of the test program. To maintain the schedule when mechanical problems occurred during testing in the CVDF, test plans were modified routinely instead of repairing the faulty components and resuming the original test procedure. This approach to testing components and subsystems individually is problematic unless a full system test is eventually completed. An integrated Bay 4 and 5 preoperational test was performed, demonstrating full operational capability prior to the Operational Readiness Reviews.

4.7 WORKER PROTECTION PROGRAM

In previous discussions with SNFP personnel, the Board's staff raised an issue involving potential radiation exposure to facility workers due to a possible spray leak from the IWTS. Similar issues existed with regard to hazards to facility workers from other potential events, such as a runaway thermal reaction. The dose to workers at 100 meters for a runaway thermal reaction in the K-Basins was calculated to be as high as 6 roentgen equivalent man cumulative effective dose equivalent. Although a definitive calculation of the dose to workers inside the K-Basin building is not required, a qualitative estimate of the potential exposure of workers would be expected as part of an activity-level hazard analysis. Since the hazard analyses reviewed by the Board's staff do not address qualitative estimates of the potential exposure of workers for these events, the staff was concerned that appropriate and adequate controls for the protection of workers in the K-W Basin had not been identified.

These concerns were discussed with representatives of the project during site visits by the Board's staff on October 3–4 and 11–12, 2000 and during a telephone conference on October 16, 2000. The issues were resolved in a November 3, 2000 DOE letter to the Board (Huntoon, 2000), as noted below.

The contractor stated that the flanged joints on the backwash piping are now fitted with lead shield blankets, which will prevent aerosol from dispersing into the air in the event of a spray leak from the joints. Although the contractor maintains his position that a rapid oxidation reaction that could result in a runaway thermal reaction in the K-W Basin is beyond extremely unlikely, certain defense-in-depth measures have been implemented to provide additional protection for the workers. Specifically, the contractor is locating additional Continuous Air Monitor alarms in the K-W Basin in work areas where potential runaway thermal reactions could occur in order to provide an early as possible warning to the workers. Training programs on uranium oxidation and consequences of a hypothetical runaway thermal reaction are being given to K-Basin workers, with emphasis on prudent measures to minimize risk in the event of a runaway thermal reaction. In addition, an emergency action plan will address recovery of operations in the event that operations are suspended due to a runaway thermal reaction.

5. FEEDBACK AND CONTINUOUS IMPROVEMENT

The work involved in the basic Integrated Safety Management function of Feedback and Improvement, as it is applied to the design and construction phase, encompasses those activities that evaluate, identify, and incorporate good practices and lessons learned during the design, procurement, installation, and testing of safety-related SSCs.

5.1 PHASED STARTUP INITIATIVE

In DNFSB/TECH-17, the Board's staff suggested that an early test of the FRS and the IWTS in the K-Basins would provide valuable information before the start of production processing. In 1999, the project established a Phased Startup Initiative (PSI) to provide limited hot operations with SNF. The objective was to provide system optimization, design verification, and an opportunity to enhance operator proficiency. Following the completion of all testing, SNF removed from canisters was to be returned to these vessels and placed back into storage racks, awaiting authorized production operations.

During 2000, extensive delays in completing preoperational testing resulted in modification of these objectives. Initial operations will involve decapping, washing, sorting, and loading of a limited number of intact SNF elements into fuel baskets in the loading queue. Once the project has completed all ORRs and hot operations have been approved, these baskets will be loaded into the first Multi-canister Overpack. Unfortunately, since the canisters being selected contain as little sludge and/or scrap as possible, the original objective of this initiative—to test the design assumptions of the FRS and IWTS—cannot be realized. The phased startup initiative was valuable in demonstrating readiness to operate these new systems, although the added value of early experience with deteriorated fuel was not achieved.

5.2 DESIGN REVIEWS

The conventional approach to a new project is to finish the preliminary design phase when approximately 30 percent of the design has been completed, and to have an in-depth design review before initiation of the final design phase and the start of construction. The preliminary design should also include a Preliminary Safety Analysis Report and an independent project cost estimate. The SNFP did not use this conventional approach, but rather, as described in Section 4.1, used a phased SAR approach in which preliminary sections of the SAR were written and approved in time to support construction of facility structures, procurement of equipment, and installation of systems in the facility. As a result, the early overview and integration of the complete project suffered, which subsequently caused additional delays and costs.

During the past 4 years, DOE-RL and the contractor, supported by numerous outside consultants and committees, conducted a total of more than 200 design and process reviews. Unfortunately, many of these mini-reviews failed to do a thorough job of identifying inadequate design features. For example, a review of the IWTS during startup testing revealed the absence of pressure relief devices in the particulate settler vessels. Considerable delay was experienced while the vessels were modified to include rupture disks. In the CVDF, the design did not call for safety-related dampers in the ventilation system. When the ventilation system was tested, the installed dampers could not maintain the required leaktightness and had to be replaced, resulting in added delay in the critical path schedule.

5.3 TEST PROGRAM EXPERIENCE

The project has experienced significant problems in transitioning from construction to operations. Difficulties in conducting construction acceptance testing and preoperational testing resulted in repeated schedule delays and delayed confirmation of the ability of the facilities to perform as designed. For example, significant delays occurred during accelerated testing of the PSI (see Section 5.1), even after facility operations commenced. Some the causes of these delays included:

- Discovery of faulty equipment installation.
- Design and/or manufacturing flaws.
- Poor construction engineering and project engineering.
- Software development issues.
- Problems with conduct of testing/operations.

Lessons learned from this testing program may not have been shared effectively across the Hanford Site and with other sites. For example, the project's preparation of test summary reports only for safety-class SSCs prior to the ORR may result in the loss of valuable information for other systems. For other than safety-class SSCs, each design authority reviews the raw test data to determine system/component fitness. This approach is likely to lead to weaknesses in the dissemination of lessons learned from the testing program because the personnel assigned to the testing organization are largely subcontractors, many with no long-term commitment to Hanford. Although test report summaries are expected to be completed for all SSCs, there is no provision for compiling and disseminating valuable input from the testing organizations.

5.4 INTEGRATED SAFETY MANAGEMENT SYSTEM VERIFICATIONS

In January 1998, DOE-RL completed a Phase I verification of the contractor's ISM System and of the K-Basins facility ISM System. On the basis of the number and extent of the gaps identified by both the contractor and the DOE ISM review team, the contractor's ISM System was not considered to be adequately institutionalized at that time.

Significant progress was made in implementing ISM in the project, and in November 1999, DOE-RL completed a combined Phase I/II ISM System verification of the K-Basins, the CVDF, and the Canister Storage Building. The Board's staff observed this verification. The verification team found that the ISM System Description for the SNFP of September 9, 1999, met the requirements of DOE Acquisition Regulations clause and DOE management direction for work activities conducted at the K-Basins. However, the team identified issues regarding implementation of ISM requirements at construction projects that needed to be addressed. In addition, the team recommended that the DOE-RL manager consider ISM to be implemented once the following items have been addressed:

- Define roles and responsibilities for safety while transitioning the construction projects to operational facilities.

- Develop and implement a Chemical Management Program.

As the project nears completion of construction and preoperational testing and prepares for operations using SNF, the ISM System is being institutionalized for the SNFP facilities. The final reports of the DOE ORR will include a statement regarding the team leader’s assessment of the adequacy of the implementation of the core functions and guiding principles of ISM addressed by the ORR.

6. CONCLUSIONS

On the basis of its reviews of design and construction activities and of the associated safety documentation, as well as extensive interactions with the staffs of DOE-Headquarters, DOE-RL, and the SNFP contractors, the Board's staff has reached the following conclusions, subject to satisfactory completion of ORRs:

1. On the basis of its review of the K-Basin SAR regarding the SNFP modifications to the K-West Basin, the Board's staff concludes that the design accounts for appropriate hazards and provides hazard controls to prevent and mitigate the identified design basis accidents (see Section 3.2). The staff believes the design features, coupled with administrative controls, provide adequate assurance that the K-West Basin can be operated in a manner that adequately protects the public, workers, and the environment from potential hazards.

2. On the basis of its review of the MCO Topical Report, the Board's staff concludes that the design of the MCO provides a robust container for extended storage of fuel in the CSB (see Section 3.3). Construction of the MCO with code stamping to the requirements of the ASME Boiler and Pressure Code, Section III, Class 1 (American Society of Mechanical Engineers, 1998), results in a level of quality equivalent to that found in pressure vessels for commercial nuclear reactors. The MCO can be used as intended in a manner that provides adequate protection of the public, workers, and the environment.

3. The Board's staff has identified no safety issues with the CTS, as described in the SARP and reviewed in the SER (see Section 3.4). Use of the CTS with robust MCOs will be a limited campaign over a restricted route. The transport time limitations associated with potential pressure buildup are addressed in Section 3.2.2.2.

4. On the basis of its review of the CVDF hazard analyses as described in the FSAR, including Annex B, and the associated SERs, the Board's staff concludes that the CVDF, as designed, can be operated in a manner that provides adequate protection of workers, the public, and the environment from potential hazards (see Section 3.5).

6. On the basis of its review of the CSB hazard analyses as described in the FSAR, including Annex A, and the associated SERs, the Board's staff concludes that the CSB, as designed, can be operated in a manner that provides adequate protection of workers, the public, and the environment from potential hazards (see Section 3.6).

7. The valuable lessons learned on this project in areas such as quality assurance, preoperational testing, phased SAR preparation, and design reviews (see Sections 4 and 5) should be applied to the ongoing project efforts for the K-East Basin. They should also be documented for application to other projects throughout DOE's defense nuclear complex.

GLOSSARY OF ABBREVIATIONS AND ACRONYMS

ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
Board	Defense Nuclear Facilities Safety Board
BPVC	(ASME) Boiler and Pressure Vessel Code
C	centigrade
CFR	Code of Federal Regulations
CLS	Cask Loadout System
CSB	Canister Storage Building
CSE	Criticality Safety Evaluation
CSSG	Criticality Safety Support Group
CTS	Cask Transportation System
CVDF	Cold Vacuum Drying Facility
DCS	distributed control system
DG	diesel generator
DOE	U.S. Department of Energy
DOE-RL	Department of Energy’s Richland Operations Office
ERO	Emergency Response Organization
F	Fahrenheit
FHA	fire hazard analysis
FRS	Fuel Retrieval System
FSAR	Final Safety Analysis Report
HEPA	high-efficiency particulate air (filter)
HVAC	heating, ventilation, and air conditioning (system)
HWVP	Hanford Waste Vitrification Project
IEEE	Institute of Electrical and Electronics Engineers
IPSS	Immersion Pail Support Structure
ISM	Integrated Safety Management
IWTS	Integrated Water Treatment System
IXM	ion exchange module
K-E	K-East (Basin)
k_{eff}	effective multiplication factor
kV	kilovolt(s)
kVA	kilovolt-amperes
kW	kilowatt
K-W	K-West (Basin)
MCNP	Monte Carlo N-Particle
MCO	Multi-canister overpack
MCS	monitoring and control system
MHM	MCO Handling Machine
NFPA	National Fire Protection Association
NPH	natural phenomena hazard

NRC	Nuclear Regulatory Commission
ORR	Operational Readiness Review
PC	Performance Category
PCM	Primary Cleaning Machine
PSI	Phased Startup Initiative
psi(g)	pounds per square inch (gauge)
Pu^{240}	plutonium-240
PUREX	Plutonium/Uranium Extraction Facility
PWCS	Process Water Conditioning System
rem	roentgen equivalent man
SAR	Safety Analysis Report
SARP	Safety Analysis Report for Packaging
SCIC	Safety-Class Instrumentation and Control (System)
SER	Safety Evaluation Report
SNF	spent nuclear fuel
SNFP	Spent Nuclear Fuel Project
SSCs	structures, systems, and components
TSR	Technical Safety Requirement
UPS	uninterruptible power supply
V	volt
VPS	Vacuum Purge System

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