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Supporting Safety Documentation for Subsurface Construction of the Canister Storage Building Below Grade Construction Restart

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SUPPORTING SAFETY DOCUMENTATION FOR
SUBSURFACE CONSTRUCTION OF THE
CANISTER STORAGE BUILDING

WHC-SD-SNF-RPT-004
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EXECUTIVE SUMMARY

The U.S. Department of Energy established the K Basins Spent Nuclear Fuel Project to address safety and environmental concerns associated with deteriorating spent nuclear fuel presently stored under water in the Hanford Site's K Basins, which are located near the Columbia River. Recommendations for a series of aggressive projects to construct and operate systems and facilities to manage the safe removal of K Basins fuel were made in WHC-EP-0830, *Hanford Spent Nuclear Fuel Recommended Path Forward*,¹ and its subsequent update, WHC-SD-SNF-SP-005, *Hanford Spent Nuclear Fuel Project Integrated Process Strategy for K Basins Fuel*². The integrated process strategy recommendations include the following process steps:

- Fuel preparation activities at the K Basins, including removing the fuel elements from their K Basin canisters, removing excess sludge from the fuel by means of flushing, as necessary, and packaging the fuel into multicanister overpacks
- Removal of free water by draining and vacuum drying at a vacuum drying facility
- Dry shipment of fuel from the vacuum drying facility to the Canister Storage Building (CSB), a new facility in the 200 East Area of the Hanford Site

¹Fulton, J. C., 1994, *Hanford Spent Nuclear Fuel Recommended Path Forward*, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

²WHC, 1995, *Hanford Spent Nuclear Fuel Project Integrated Process Strategy for K Basins Fuel*, WHC-SD-SNF-SP-005, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

- Temporary staging in the CSB of multicannister overpacks containing partially dried fuel
- Removal of the remaining absorbed water from the multicannister overpacks at a conditioning facility
- Interim dry storage of the multicannister overpacks in the CSB until a suitable long-term repository is established.

The CSB originally was designed for storage of the glass canisters for the Hanford Waste Vitrification Plant. Construction of the vault structure for the CSB was initiated before the termination of the Hanford Waste Vitrification Plant Project. The base slab and portions of the exterior walls of the vault that were completed before termination of the Hanford Waste Vitrification Plant Project are incorporated into the CSB design for the Spent Nuclear Fuel Project.

The purpose of this report is to provide an evaluation of the facility design criteria, the design compliance with the applicable criteria, and the basis for authorization to proceed with construction of the CSB below-grade structure. The evaluation is based on the format of DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*.³ The evaluation includes those sections of DOE-STD-3009-94 that are applicable to the CSB below-grade structure based on agreement between the U.S. Department of Energy, Richland Operations Office, and Westinghouse Hanford Company. The information provided is based on the CSB design information currently available. The information includes the

³DOE-STD-3009-94, 1994, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, U.S. Department of Energy, Washington, D.C.

subject chapter, a summary of reference information available that documents compliance or the intent to comply with the applicable criteria, and an evaluation section that identifies the degree to which compliance is demonstrated.

This evaluation identified the following items that could impact the below-grade structure.

1. Westinghouse Hanford Company safety class 1 items are required to be protected from tornado missiles to meet the U.S. Nuclear Regulatory Commission equivalency policy established for the Spent Nuclear Fuel Project. Tornado protection will require hardening the local confinement barriers, hardening the superstructure, or establishing a probabilistic risk approach to reduce the requirement for tornado-resistant safety class items. Hardening of the CSB superstructure and the inlet and exhaust plenums for protection from tornado missiles could increase the loading on the below-grade structure. The possibility of increased loading has not been considered in the structural evaluation of the below-grade structure. The design approach for tornado protection will be developed based on minimizing impact on below-grade structures.
2. An assessment of the condition of the existing basemat and walls is in progress to identify any degradation of structural conditions or materials that may have occurred since construction was terminated in October 1993. This assessment and any necessary repair work will ensure that the substructure will meet applicable criteria for the safety class 1 structure.

3. Hot conditioning is not addressed in this report. It will be performed in a facility incorporated into an extension to the south end of the building. Structural design of this extension will be initiated shortly. Feasibility studies have concluded that this extension will need to be seismically decoupled from the existing vault to avoid impacting the design schedule and construction start.

Subject to satisfactory closure of the above items, the evaluation demonstrates that the design for the below-grade structure will meet the applicable safety-related criteria and that appropriate design features have been provided to prevent potential accidents that could result in undue risk to the public, the workers or the environment.

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LIST OF TERMS

ARF	airborne release fraction
CSB	Canister Storage Building
DBA	design basis accident
DBE	design basis earthquake
DOE	U.S. Department of Energy
ICF KH	ICF Kaiser Hanford
ISFSI	independent spent fuel storage installations
HEPA	high-efficiency particulate air (filter)
HWVP	Hanford Waste Vitrification Plant
MCO	multicanister overpack
MHM	multicanister overpack handling machine
MSL	mean sea level
NCR -	nonconformance report
NPH	natural phenomena hazard
NRC	U.S. Nuclear Regulatory Commission
PMP	probable maximum precipitation
PHA	preliminary hazards analysis
QAPP	quality assurance program plan
RF	respirable fraction
SNF	spent nuclear fuel
SSC	structure, system, and component

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1.0 SITE CHARACTERISTICS

1.2 REQUIREMENTS

This section lists the design codes, standards, regulations, and U.S. Department of Energy (DOE) orders that are required for establishing the safety basis for the below-grade portion of the Canister Storage Building (CSB). Only the requirements that are specific to Chapter 1.0 and that pertain to the safety analysis are provided.

1.2.1 U.S. Department of Energy Orders and Standards

The following DOE orders and standards are applicable to the safety analysis.

- DOE Order 6430.1A, 1989, *General Design Criteria*. General siting requirements for meteorology, hydrology, and seismology are established in Division 1, "General Requirements," Section 0111-99.0, "Nonreactor Nuclear Facilities - General," and in Division 2, "Site and Civil Engineering," Section 0200-99.0, "Nonreactor Nuclear Facilities - General of the Order." DOE Order 6430.1A is to be replaced by DOE Order 420.1, *Facility Safety*.
- DOE Order 5480.28, 1993, *Natural Phenomena Hazards Mitigation*. This Order establishes DOE facility safety requirements for dealing with natural phenomena hazards (NPHs). The Order establishes NPH mitigation requirements, performance categories, and target probabilistic performance goals for each category. DOE Order 5480.28 will be replaced by DOE Order 420.1, which is expected to have the same NPH requirements as the implementing standards listed below.
- DOE-STD-1020-94, 1994, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*. This standard uses overall facility hazard classifications to assign NPH performance categories.
- DOE-STD-1021-93, 1993, *Natural Phenomena Hazards Performance Categories Guidelines for Structures, Systems, and Components*. This standard provides guidance for assigning structures, systems, and components (SSCs) to one of five performance categories and recommends systematic procedures for implementing this guidance.
- DOE-STD-1022-94, 1994, *Natural Phenomena Hazards Site Characteristics Criteria*. This standard provides criteria for selecting the site-specific information needed to implement DOE Order 5480.28.

- DOE-STD-1023-95, 1995, *Natural Phenomena Hazards Assessment Criteria*. This standard provides criteria for assessing NPHs to ensure that adequate design basis load levels are established for design or evaluation of DOE facilities.
- DOE-STD-1027-92, 1992, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*. This standard is used by DOE-STD-1021-93 to evaluate facility hazard classification.

1.2.2 U.S. Nuclear Regulatory Commission Rules and Guidance

The following U.S. Nuclear Regulatory Commission (NRC) rules and guidance provide input to the safety analysis.

- 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," *Code of Federal Regulations*. This rule provides the siting criteria and NPH considerations used to license independent spent fuel storage installations (ISFSIs).
- Regulatory Guide 1.76, 1974, *Design Basis Tornado for Nuclear Power Plants*. This guide provides design basis tornado wind speeds, pressure drops, and rate of pressure drops for the three regions into which the NRC divides the United States.
- NUREG-0800, 1981, *Standard Review Plan*. Standard Review Plans 2.4.1, "Hydrologic Description," 2.4.2, "Floods," 2.4.3, "Probable Maximum Flood (PMF) on Streams and Rivers," 2.4.2, "Potential Dam Failures," and 2.4.10, "Flood Protection Requirements," provide NRC guidance relative to flood protection. Standard Review Plan 3.5.1.4, "Missiles Generated by Natural Phenomena," provides guidance on the standard sets of tornado-generated missiles for use as design basis input for the various tornado regions defined in Regulatory Guide 1.76 (NRC 1974).
- SECY-93-087, 1993, *Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs*. This paper provides information regarding NRC staff-proposed reductions of tornado wind speeds relative to NPH. These staff recommendations have not yet been accepted by the Commission for generic application although they have been applied on a facility-specific basis (Clifford 1996).

1.2.3 Industry Consensus Standards and Other Documentation

The following industry standards are applicable to the safety analysis.

- ANSI/ANS-2.8-1992, 1992, *Determining Design Basis Flooding at Power Reactor Sites*. This standard provides criteria to establish design basis flooding for nuclear safety-related features at power reactor sites.

- ASCE, 1961, "Wind Forces on Structures," *Transaction of the American Society of Civil Engineers*, Volume 126, Part II. This standard establishes the methodology for straight wind calculations.
- ASCE 7-93, 1993, *Minimum Design Loads for Building and Other Structures*. This standard sets the minimum acceptable fastest mile wind speed.
- ICBO, 1994, *Uniform Building Code*. The Uniform Building Code is used for the seismic design of safety class 2 SSCs.
- NFPA 780, 1995, *Lightning Protection Systems*. This code provides requirements for the safeguarding of persons and property from hazards arising from exposure to lightning.

1.4.2 Hydrology

The purpose of this section is to provide the hydrological information necessary to understand any regional hydrological phenomena of concern for facility operation and to understand any dispersion analyses performed.

The hydrology design requirements for the CSB are discussed in detail in WHC-SD-SNF-DB-009, *Canister Storage Building Natural Phenomena Hazards* (WHC 1996a, pp 14-16). The following discussion of hydrological design requirements is a summary of the information provided in that document.

1.4.2.1 Stream and River Flooding. The CSB site is at an elevation approximately 213 m (700 ft) above mean sea level (MSL). The Columbia River is closest to the CSB at river kilometer 595 (mile 370) where the normal flow is approximately 115 m (375 ft) above MSL. At the same river kilometer, the estimated flood level at a mean annual probability of 1×10^{-4} is 133 m (435 ft) above MSL (McCann and Boissonnade 1988). This is about 80 m (265 ft) below the CSB. Cold Creek and Dry Creek are ephemeral streams on the western portion of the Hanford Site. They are not of concern for the CSB site as these streams are in the Yakima River drainage basin, and the probable maximum flood is approximately 30 m (100 ft) below the elevation of the divide between the Columbia River drainage basin, where the CSB is located, and the Yakima River drainage basin.

The considerations listed above make the CSB site a flood-dry site with respect to river flooding. As such, no additional information on stream and river flooding is required to address the safety of the facility. Information on the flooding histories of the Columbia River, the Yakima River, and the Cold Creek watersheds are provided in WHC-SD-W058-PSAR-001, *Replacement of the Cross-Site Transfer System, Preliminary Safety Analysis Report* (WHC 1996b, Section 5.4).

1.4.2.2 Probable Maximum Precipitation. Guidance in DOE-STD-1020-94 states that the performance goals for SSCs must be satisfied when subjected to the design basis flood level for local precipitation. This can be accomplished by grading and draining the site to provide sufficient runoff capacity such that the facility is not challenged, or by designing facility features to

accommodate the water level. The performance goal is 1×10^{-4} for a Performance Category 3 structure such as the CSB.

A recent cooperative study by the National Oceanic and Atmospheric Administration, the Bureau of Reclamation, and the Army Corps of Engineers has updated the probable maximum precipitation (PMP) estimates for the Pacific Northwest (Hansen et al. 1994). This document supersedes earlier work by these organizations and is the source used for the PMP shown in Table 1.4-1. The PMP values are estimates of the maximum precipitation physically possible for both general storms (large air mass interactions) and local storms (unstable air, thunderstorms). At the Hanford Site, the local storm produces more precipitation than the general storm. No annual probability of exceedance is given in Hydrometeorological Report No. 57, *Probable Maximum Precipitation - Pacific Northwest States* (Hansen et al. 1994) for the PMP for either general or local storms. The PMP is generally judged to have an annual probability of exceedance of 1×10^{-6} or less. If the 6-hour local storm has an annual probability of 1×10^{-6} , the PMP values for an annual probability of 1×10^{-4} are extrapolated to be approximately 13 cm (5 in.) when considering a 3-km² (1-mi²) area and approximately 9 cm (3.5 in.) when considering a 26-km² (10-mi²) area.

NRC requirements for flood protection for ISFSIs were also considered in WHC-SD-SNF-DB-009 (WHC 1996, p 16). NRC guidance in 10 CFR 72.90(f), "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," states that "the facility must be sited so as to avoid to the extent possible the long-term and short-term adverse impacts associated with the occupancy and modification of floodplains." No guidance is given for the probabilities of flooding or precipitation. For nuclear power reactors, the 24-hour PMP estimates were recommended for use in constructing runoff models to estimate the discharge to the site's storm drainage system (NRC 1981, ANSI/ANS 1992). At the Hanford Site, using the local storm is more conservative than using the general storm, which occurs over a larger area. Thus, the local 3-km² (1-mi²), 6-hour storm rate of 23 cm (9.2 in.) is specified for the CSB (Table 1.4-1). The slightly lower 26-km² (10-mi²) rate of 19 cm (7.4 in.) can be used for calculating the surrounding watershed runoff.

The mean annual precipitation for the Hanford Site is 16 cm (6.3 in.). The historical record indicates the maximum amount of precipitation ever recorded on the Hanford Site in a 24-hour period was 5 cm (1.9 in.).

Because the CSB will be located atop the 200 Area plateau, even the PMP values of 23 cm (9.2 in.) and 19 cm (7.4 in.) in 6 hours will not threaten the building as much of the water will infiltrate the soil and the remainder will drain towards the Columbia River. Local protection for runoff will be provided by earthwork and grading design, and by the storm drain system consisting of pipes, culverts, ditches, channels, and catch basins, as required.

Table 1.4-1. Probable Maximum Precipitation Design Loads for the Canister Storage Building.

	PMP 24-hour general storm 26 km^2 $(10 \text{ mi}^2)^a$	PMP local storm 3 km^2 $(1 \text{ mi}^2)^a$	PMP local storm 26 km^2 $(10 \text{ mi}^2)^a$	25-year average return period ^b	100-year average return period ^b	1,000-year average return period ^b
15 minutes	--	4.0	3.2	--	--	--
20 minutes	--	--	--	0.47	0.60	0.80
30 minutes	--	6.0	4.8	--	--	--
45 minutes	--	7.2	5.8	--	--	--
1 hour	1.6	8.0	6.4	0.62	0.81	1.11
6 hours	4.7	9.2	7.4	1.21	1.59	2.20
24 hours	8.0	--	--	1.56	1.99	2.68
48 hours	9.6	--	--	--	--	--
72 hours	10.4	--	--	--	--	--

Notes: This table is taken from WHC, 1996, Canister Storage Building Natural Phenomena Hazards, WHC-SD-SNF-DB-009, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
Precipitation depths are in inches; to convert to centimeters, multiply by 2.54.

^aHansen, E. M., D. D. Fenn, P. Corrigan, Vogel, L. C. Schreiner, and R. W. Stodt, 1994, Probable Maximum Precipitation - Pacific Northwest States, Hydrometeorological Report No. 57, National Weather Service, Silver Spring, Maryland.
^bStone, W. A., J. M. Thorp, O. P. Gifford, and D. J. Hollink, 1983, Climatological Summary for the Hanford Site, PNL-4622, Pacific Northwest Laboratory, Richland, Washington.
PMP = Probable maximum precipitation
-- = No data available.

1.4.2.3 Subsurface Hydrology. Figure 1.4-1 shows the water table for the Hanford Site. For the 200 East Area the figure shows the water table to be about 126 m (413 ft) above MSL. The CSB site is about 213 m (700 ft) above MSL, and the foundation is 201 m (662 ft) above MSL. Thus, with 87 m (287 ft) between the surface of the site and the water table and 75 m (149 ft) between the foundation and the water table, subsurface hydrology has no significant impact on the structural design of the CSB.

No water withdrawals from or direct discharges to the water table are planned, and no liquid effluent waste streams will be generated.

Additional information on the subsurface hydrology for the Hanford Site is provided in WHC-SD-W058-PSAR-001 (WHC 1996b, Section 5.4.3).

1.4.3 Geology

The purpose of this section is to provide the geologic information necessary to understand any regional geologic phenomena of concern for facility operation.

The Hanford Site lies in the Pasco Basin near the eastern limit of the Yakima Fold Belt. The Site is underlain by basalt of the Columbia River Basalt Group, which is covered by about 122 m (400 ft) of sediment at the CSB site. The Site is in an area of low magnitude seismicity under north-south compressional stress, which is reflected in the deformation of the Yakima Ridges. The major contributors to the seismic hazard in and around the Hanford Site are (1) fault sources related to the Yakima Folds, (2) shallow basalt sources that account for the observed seismicity within the Columbia River Basalt Group and not associated with the Yakima Folds, (3) crystalline basement source region, and (4) Cascadia Subduction Zone earthquakes.

The DOE seismic design requirement for nuclear facilities is provided in DOE Order 5480.28, which, along with its implementing standards, uses a graded approach in applying seismic design criteria. Facility SSCs are assigned one of five performance categories in accordance with the guidance provided in DOE-STD-1021-93. The performance categories provide a graded application of seismic design criteria for selecting the design basis earthquake (DBE) and structural design rules based on consideration of the relative probabilities and consequences of potential damage or failure of a facility.

Figure 1.4-1. Water Table Map for the Hanford Site.

Each performance category has an associated NPH performance goal with a mean annual probability of exceedance of acceptable behavior limits. The NPH performance goal is a measure of the level of protection against potential seismic events and is used as a target for the establishment of seismic mitigation design requirements.

Performance goals and related seismic hazard probabilities are presented in Table 1.4-2.

Table 1.4-2. Seismic Performance Goals and Specific Seismic Hazard Probabilities.

P_F (1/year)	Performance category	P_H (1/year)	Return period (yr)	Risk reduction factor ($R_R = P_H/P_F$)
1×10^{-3}	PC-1	2×10^{-3}	500	2
5×10^{-4}	PC-2	1×10^{-3}	1,000	2
1×10^{-4}	PC-3	5×10^{-4}	2,000	5
1×10^{-5}	PC-4	1×10^{-4}	10,000	10

Note: This information taken from DOE Order 5480.28, Natural Phenomena Hazards Mitigation, U.S. Department of Energy, Washington, D.C., and DOE-STD-1020-94, Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities, U.S. Department of Energy, Washington, D.C.

P_F = Performance goal (annual probability of exceedance of acceptable behavior limits).
 P_H = Seismic hazards exceedance probability (return period = $1/P_H$).

The CSB (including the MCOs, storage tubes, vault substructure, and operating deck) has been classified as NPH Performance Category 3 and will be designed to Performance Category 3 requirements. Performance Category 3 is anchored at the 5×10^{-4} seismic hazards exceedance probability (2,000-year return period) for ground motion (see Table 1.4-2). Based on the probabilistic seismic hazard assessment of the Hanford Site (Youngs 1996), the 2,000-year peak horizontal ground motion is approximately 0.24g at the CSB site.

The current CSB was designed to a Newmark and Hall median horizontal seismic design spectra anchored at 0.35g (Newmark and Hall 1978). The design was developed in anticipation of the draft DOE-STD-1020-94 and the draft seismic hazard assessment of the Hanford Site (WHC 1994). The return period for 0.35g is about 8,000 years (Youngs 1996). The 5% damped, equal-hazard, horizontal and vertical response spectra for the current CSB are included in WHC-SD-SNF-DB-004, *Spent Nuclear Fuel Project Seismic Design Criteria* (WHC 1995).

The response spectra illustrate that the current CSB seismic design response spectra conservatively envelop the Performance Category 3 response spectra over the entire range of frequencies for both vertical and horizontal response spectra. The CSB design methodology envelops the requirements of DOE Order 5480.28 and DOE-STD-1020-94.

Additional information on the geology and seismology for the Hanford Site is provided in WHC-SD-W058-PSAR-001 (WHC 1996b, Section 5.5).

1.5 NATURAL PHENOMENA THREATS

The purpose of this section is to provide identification of specific natural phenomena events, such as DBEs, considered to be potential accident initiators. Assumptions supporting Chapter 3.0, "Hazard and Accident Analysis," will be summarized.

The NPH design requirements for the CSB are discussed in detail in WHC-SD-SNF-DB-009 (WHC 1996, entire document). Table 1.5-1 of this report is a summary of the NPH design requirements taken from that reference.

Safety class 1 SSCs are required to be designed to withstand the most severe design basis NPH loadings. The below-grade portion of the CSB has been assigned the most conservative safety classification possible, safety class 1, so its design reflects the ability to withstand all design basis NPHs. Analyses of design basis events, therefore, will have no impact on the below-grade portion of the CSB's design and construction.

Beyond design basis analyses performed to identify cost-effective actions that could be taken to reduce residual risk will be documented in the safety analysis report. These beyond design basis analyses will consider beyond design basis seismic events. In addition, consideration will be given to the cumulative effect that failure of SSCs not designed for a particular NPH may have on the facility.

Table 1.5-1. Natural Phenomena Loads for Safety Class 1 Aspects of the Canister Storage Building.

Hazard	Load	Design guidance
Seismic	Median response spectra: ^a 0.35g horizontal 0.23g vertical	DOE Order 5480.28 ^b DOE Standard 1020-94 ^c
Straight wind	129 km/h (80 mi/h), fastest mile at 9 m (30 ft)	ASCE-7 ^d DOE Standard 1020-94 (including missiles)
Tornado	Wind speeds 322 km/h (200 mi/h) total 257 km/h (160 mi/h) rotational 64 km/h (40 mi/h) translational	NRC Standard Review Plan ^e 3.3.2 Tornado Loading 3.5.1.4 Missiles Generated by Natural Phenomena NRC SECY-93-087 ^f
Volcanic ash	117 kg/m ² (24 lb/ft ²) ground ash load	SDC 4.1, Rev. 12, for ash load combinations ^g
Flooding	Dry site for river flooding Site drainage basin: 19 cm (7.4 in.) for 6-hour probable maximum precipitation Site drainage: 23 cm (9.2 in.) for 6-hour probable maximum precipitation	ANSI/ANS-2.8-1992 ^h NRC Regulatory Guide 1.59 ⁱ NRC Standard Review Plan ^f 2.4.2 Floods
Lightning	Lightning protection shall be considered for facility.	NFPA 780 ^j
Snow	98 kg/m ² (20 lb/ft ²) ground load	ASCE-7 ^e

Note: This table is taken from WHC, 1996, Canister Storage Building Natural Phenomena Hazards, WHC-SD-SNF-DB-009, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

^aNewmark, N. M., and W. J. Hall, 1978, Development of Criteria for Seismic Review of Selected Nuclear Power Plants, NUREG/CR-0098, U.S. Nuclear Regulatory Commission, Washington, D.C.

^bDOE Order 5480.28, Natural Phenomena Hazards Mitigation, U.S. Department of Energy, Washington, D.C.

^cDOE-STD-1020-94, Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities, U.S. Department of Energy, Washington, D.C.

^dASCE, 1988, Minimum Design Loads for Building and Other Structures, ASCE 7, American Society of Civil Engineers, New York, New York.

^eNRC, 1981, Standard Review Plan, NUREG-0800, U.S. Nuclear Regulatory Commission, Washington, D.C.

^fNRC, 1993, Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs, SECY-93-087, U.S. Nuclear Regulatory Commission, Washington, D.C.

^gSDC 4.1, 1993, "Standard ARCH-CIVIL Design Criteria for Design Loads for Facilities," Hanford Plant Standards Design Criteria, HPSDCM, Rev. 12, Westinghouse Hanford Company, Richland, Washington.

^hANSI/ANS, 1992, Determining Design Basis Flooding at Power Reactor Sites, ANSI/ANS-2.8-1992, American Nuclear Society, La Grange Park, Illinois.

ⁱNRC, 1977, Design Basis Floods for Nuclear Power Plants, Regulatory Guide 1.59, U.S. Nuclear Regulatory Commission, Washington, D.C.

^jNFPA, 1992, Lightning Protection Code, NFPA 780, National Fire Protection Association, Quincy, Massachusetts.

1.9 REFERENCES

- Clifford, J. W., 1996, *Revision of Tornado Design Criteria for the Washington Public Power Supply System (WPPSS) Nuclear Project No. 2 (WNP-2) (TAC No. M88626)* (letter GI2-96-032 to J. V. Parrish, Washington Public Power Supply System, January 24), U.S. Nuclear Regulatory Commission, Washington, D.C.
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- Newmark, N. M., and W. J. Hall, 1978, *Development of Criteria for Seismic Review of Selected Nuclear Power Plants*, NUREG/CR-0098, U.S. Nuclear Regulatory Commission, Washington, D.C.
- NRC, 1974, *Design Basis Tornado for Nuclear Power Plants*, Regulatory Guide 1.76, U.S. Nuclear Regulatory Commission, Washington, D.C.
- WHC, 1994, *Probabilistic Seismic Hazard Analysis, DOE Hanford Site, Washington*, WHC-SD-W236A-TI-002, Rev. 0, prepared for Westinghouse Hanford Company, Richland, Washington, by Geomatrix Consultants, San Francisco, California.
- WHC, 1995, *Spent Nuclear Fuel Project Seismic Design Criteria*, WHC-SD-SNF-DB-004, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1996a, *Canister Storage Building Natural Phenomena Hazards*, WHC-SD-SNF-DB-009, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1996b, *Replacement of the Cross-Site Transfer System, Preliminary Safety Analysis Report*, WHC-SD-W058-PSAR-001, Rev. 1, Westinghouse Hanford Company, Richland, Washington.
- Youngs, R. R., 1996, *Revised Seismic Hazard Report for Hanford DOE Sites* (letter to A. M. Tallman, February 12), Geomatrix Consultants, Incorporated, San Francisco, California.

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2.0 FACILITY DESCRIPTION

2.2 REQUIREMENTS

This section lists the design codes, standards, regulations, and DOE orders that are required for establishing the safety basis of the facility. The intent is to provide only the requirements that are specific for this chapter and pertinent to the safety analysis.

2.2.1 U.S. Department of Energy Orders and Standards

The following DOE orders and standards are applicable to the safety analysis.

- Doe Order 6430.1A, 1989, *General Design Criteria*. The main reference standards and guides for facility design are presented in Division 13, "Special Facilities." Section 1320, "Irradiated Fissile Material Storage Facilities," contains additional requirements for irradiated fissile material storage facilities. In particular, paragraph 1320-4, "Special Design Features," requires that the passive cooling air system that ensures an acceptable temperature for the stored material be designed as a safety class system. The structures that provide natural convective cooling air flows for the MCOs are designed to safety class 1 criteria.
- DOE-RL-HPS-SDC-4.1, 1993, *Standard Architectural-Civil Design Criteria, Design Loads for Facilities*. This Hanford Site plant standard provides minimum requirements for the design of the below-grade portion of the CSB. As stated in WHC-SD-HWV-PSE-001, *Hanford Waste Vitrification Plant Canister Storage Building Preliminary Safety Analysis Report Addendum* (WHC 1994), this is one of the main reference standards for facility design. Section 3.1 and Section 3.3 of this standard provide design criteria loads and load combinations for safety class 1 SSCs. These include nominal dead, live, snow, and soil loads; normal operating loads; and natural phenomena loading of extreme wind, earthquake, ashfall, and flood.

Natural phenomena loads listed on Table 1.5-1 of Section 1.5, "Natural Phenomena Threats," will be analyzed. For example, the tornado wind load from Table 1.5-1 is given as 322 km/h (200 mi/h) total, versus the SDC-4.1 wind load of 145 km/h (90 mi/h). This higher wind load on the superstructure will be incorporated into the appropriate analysis load combination.

2.2.2 U.S. Nuclear Regulatory Commission Rules and Regulations

The following NRC rule and guidance are applicable to the safety analysis.

- ~ 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Subpart F, "General Design Criteria," and Subpart H, "Physical Protection," *Code of Federal Regulations*. This rule provides general design criteria for a spent fuel storage facility. Select sections of the rule have been invoked by the Regulatory Requirements Team as a means to comply with NRC equivalency criteria. These criteria are given in WHC-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements* (WHC 1995c). The design will incorporate these additional criteria applicable to the CSB.

2.2.3 Industry Consensus Standards and Other Documentation

The following industry standards are applicable to the safety analysis.

- ~ ANSI/ANS-57.9-1992, 1992, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)*. This standard includes requirements for the design of major buildings and structures including physical security features for ISFSIs. The architect-engineer has performed a review of the performance and design requirements and has concluded that the current Spent Nuclear Fuel (SNF) CSB design appears to be in general compliance with the exception of transportation package washdown, decontamination facilities, and safety class electrical power. Section 6.17.1.1 of this standard also requires that the most adverse dead load loading condition be increased by 5% from the estimated value. An increase in the dead load loading condition and other criteria will be evaluated in accordance with NRC equivalency policy.

2.4 FACILITY STRUCTURE

2.4.1 Facility Overview

A conceptual design for the CSB was evaluated and described in WHC-SD-W379-CDR-001, *Spent Nuclear Fuel Canister Storage Building Conceptual Design Report* (WHC 1995b). The report concluded that a facility design that takes advantage of the existing Hanford Waste Vitrifaction Plant (HWVP) CSB infrastructure and associated design basis documentation would adequately fulfill the SNF staging and storage mission. The design has continued to evolve through engineering studies and advanced conceptual design.

The SNF CSB consists of three equally sized below-grade concrete vaults covered by a concrete operating deck. Only the northernmost vault (vault 1) will be equipped with tubes to provide storage for MCOs. The storage tubes are supported from the base slab of the vault and are accessed through shield plugs in the operating deck. The operating deck and other operations areas

are enclosed in a steel building. The MCO storage vault is cooled by natural convection through dedicated inlet and exhaust air stacks and plenums. Support functions and equipment are housed in a smaller building at the north side of the operations building.

MCOs will be transported from the K Basins in shielded casks and received into the operating area at the northwest corner of the CSB. A gantry crane is used to transfer the casks from the truck to a below-grade service pit at the north end of the operating area. MCOs are removed from the cask and placed into tubes in the operating deck by the MCO handling machine (MHM).

The portion of the SNF CSB that is covered by this safety documentation submittal is referred to as the "vault" or "below-grade" package. The below-grade package includes only heavy, thick concrete structures: the interior and exterior vault walls, the intake and exhaust plenums, and the base slab. Not included in the vault construction package are the operating deck, tubes, plugs, MHM, gantry crane, service pit, support building, above-grade metal building superstructure, inlet stacks, exhaust stacks, or truck bay. The below-grade construction ends at the point where the deck begins at elevation 215 m (705 ft) (Drawings H-2-117795 and H-2-117798).

The safety classification of the CSB vault structure is covered on pages 3-7 and 3-8 of WHC-SD-HWV-PSE-001 (WHC 1994), which designated all of the structures contained in the present below-grade or vault package as safety class 1. Portions of the vault that were completed prior to suspension of the HWVP project in 1993 are incorporated into the present design. These include the base slab and portions of the walls.

The current SNF CSB vault is structurally similar to the HWVP CSB vault with a few exceptions. Architectural renderings of the present and former below-grade CSB designs are attached to promote understanding of the differences (see Architectural Renderings). These differences are discussed below.

First, each of the three vault areas are now physically separate instead of sharing common intake and exhaust plenums. This increases flexibility of use and improves the convective cooling. The convective cooling was satisfactory in the HWVP CSB with tubes containing canisters of vitrified high level waste. The heat loading density for one vault containing SNF MCOs is less than it would have been with the high-level waste canisters, hence, lower vault temperatures. This allows elimination of the requirement for insulating concrete in vault 1.

The physical separation is achieved by the extension of the interior walls to create separate vault areas. This modification contributes to reducing the radiation streaming around the air intake stack area. With partial interior walls in the HWVP design this streaming was mitigated through the use of a set of heavy concrete louvers. The reduced source term in the SNF CSB design allows for removal of the louvers from the design. This design revision reduces the construction complexity and results in savings as compared to the HWVP CSB.

Fully isolating vault 1 from the other vaults provides sufficient shielding to permit personnel to install tubes and plugs in vaults 2 and 3 at a future time. Shielding analysis assumes that vault 1 is homogeneously loaded with maximum source term MCOs. This change eliminates the need to completely furnish all three vaults with tubes and plugs before placing MCOs in the CSB. This provides for increased flexibility, as compared to the HWVP CSB design, and results in a significant near-term construction time and cost savings as well as a capital cost deferral until missions are approved for the other two vaults.

2.4.2 Facility Description

As described in Section 3.0 and as shown on Figures 3-1 through 3-9 of the WHC-SD-HWV-PSE-001 (WHC 1994), the HWVP CSB consists of a below-grade, standard, reinforced concrete structure, with approximate dimensions of 55 m x 50 m x 15 m deep (180 ft x 165 ft x 48 ft deep), and an above-grade steel operating area structure. The reference elevation of the facilities at the top of the operating deck is 216 m (709 ft) above MSL. The basemat is nominally 1.7 m (5 ft-6 in.) thick; its surface elevation is 203.2 m (666 ft-9 in.). The distance from the basemat floor to the underside of the operating deck is 11.4 m (37 ft-3 in.). The exterior walls and air inlet and outlet plenums are 1.4 m (4 ft-6 in.) thick. Interior partition walls between vaults are 0.9 m (3 ft) thick.

The HWVP CSB design as described in Section 3.0 and as shown in Figures 3-1 through 3-9 of the WHC-SD-HWV-PSE-001 (WHC 1994) contained two air intake towers and one exhaust stack providing natural convective cooling. Common below-grade intake and exhaust plenums served all three sections of one large vault. The SNF CSB configuration consists of three equally sized below-grade reinforced concrete vaults covered by a concrete operating deck.

Although not part of the vault package, a 1.5-m- (5-ft-) thick reinforced concrete operating deck will eventually form the at-grade portion of the CSB. The basemat contains embeds that will serve as alignment and horizontal seismic restrain points for the storage tubes.

Only one of the vaults will be completed with carbon steel storage tubes and individual intake and exhaust stacks at this time. This will expedite the receipt of SNF containers. The vault to be completed, the northernmost of the three, is referred to as vault 1. The remaining two vaults are to be completed to the deck level only but will remain empty for an as yet undefined future use. It is anticipated that the other two vaults may contain SNF, cesium or strontium capsules, vitrified high-level waste, or other spent fuel materials.

The below-grade walls and basemat carry the loads associated with handling and transporting the MCO (i.e., the MCO cask transporter, the cask gantry crane, and the operating area crane). As described further in Section 3.4, "Hazards Analysis," of this document, hazards analyses have determined that the below-grade portion of the CSB is safety class 1.

2.4.3 Facility Mission

Vault 1 will contain 220 storage tubes, each capable of staging or storing two MCOs, plus five additional locations to accommodate future MCO overpacks, one MCO per overpack tube, and one MCO overpack dedicated for impact limiter storage. Each storage tube will contain an impact limiter to mitigate the consequences of a dropped MCO. Initially the MCOs containing SNF will be staged in a vented but inerted condition in the storage tubes. Eventually the MCOs containing SNF will be reprocessed in a Hot Vacuum Conditioning Facility. Upon completion of hot vacuum conditioning, the MCOs will be stored in a sealed configuration, as described in Section 3.1.1 of WHC-S-0425, *Performance Specification for the Spent Nuclear Fuel Canister Storage Building* (WHC 1995b).

2.4.4 Facility Design Assumptions

Shielding calculations have assumed that each MCO contains the maximum activity value associated with MKIV fuel listed in Table 3.2.2.1.2.2-1 and described in Section 3.2.2.1.6 of WHC-S-0425 (WHC 1995b). Vault thermal analyses have assumed that 20% of the MCOs contain the maximum heat value associated with MKIV fuel, 80% the average value, as directed by WHC in WHC-S-0425 (WHC 1995b), Section 3.2.2.1.2.2, "Design Feed," and as given in Table 3.2.2.1.2.2-2, "MCO General Attributes."

2.4.5 Facility Modifications to Hanford Waste Vitrification Plant Canister Storage Building

The conversion of the CSB from the HWVP mission to the current SNF mission resulted in two structural revisions caused by the lower radiation source term and lower heat load associated with the SNF MCOs as compared to the HWVP canisters. One is the removal of the intake plenum louvers, the other is the deletion of insulating concrete on the interior walls and underside of the operating deck.

The total activity is given as 310,798 Ci for one MCO versus 891,000 Ci (5.4×10^5 Ci/gal $\times$ 165 gal) for the vitrified waste canisters. The new source term is driven primarily by alpha emitters as compared with gamma emissions from HWVP canisters. This information can be obtained from Table 3.2.2.1.2.2-1 of WHC-S-0425 (WHC 1995b) and Table 8.2-2 of WHC-SD-HWV-PSAR-001, *Hanford Waste Vitrification Plant Preliminary Safety Analysis Report* (WHC 1992).

In Section 3.2.2.1.2.2 of WHC-S-0425 (WHC 1995b) and page 3-32 of the WHC-SD-HWV-PSE-001 (WHC 1994), the heat loads for the HWVP canisters are given as 800 canisters at 0.4 kW and 1,200 canisters at 1 kW, and for the SNF MCOs as 80% at 401 W and 20% at 852 W.

The CSB design for the HWVP and SNF mission, as shown in Figures 3-1 through 3-9 in WHC-SD-HWV-PSE-001 (WHC 1994), also had partial-length, below-grade interior walls and common intake and exhaust plenums. The current CSB

configuration extends the below-grade walls to isolate the individual vaults and creates a separate intake and exhaust plenum for vault 1. This design change also aided in the deletion of the intake shielding louvers.

The below-grade portion of the HWVP CSB was designed to carry the loads associated with a large, wheeled, shielded canister transporter that was to be used for the HWVP mission. The current CSB configuration uses a large heavy crane, the MHM, to transport the MCOs. Details of the MHM and other MCO handling equipment are given in Section 2.4.3.3.7 of WHC-S-0425 (WHC 1995b).

DOE 6430.1A, Section 1300-3.2, considers safety class items as those whose failure would result in exposures exceeding DOE 5400 series limits for the public at the Site boundary or nearest point of public access. For the Hanford Site's 200 Area, this distance is 15,000 m.

--The NRC offsite boundary criteria require that there be no measurable radioactive releases from the CSB at the 100 m controlled area boundary. The impact on the CSB design would likely be an increase in the number of safety class items associated with maintaining confinement of the MCOs. Safety class items are required to withstand NPHs including tornado wind loads and missiles.

NPH wind loads and missiles associated with a tornado will be accommodated by the design. Safeguarding against tornado-driven missiles will require hardening the local confinement barriers at the MCO service pit and on the MHM or hardening the superstructure. Alternately, using a probabilistic risk assessment approach may reduce the requirement for tornado-resistant SSCs. Hardening of the superstructure could increase the loading on the below-grade walls and foundation. The impact associated with increased loading to the below-grade structure will be evaluated as part of the decision process. The decision on the design approach will be made during definitive design.

2.4.6 Construction Restart Issues

A readiness for construction task force of key project participants has identified open issues that need to be addressed before the restart of construction. These issues are documented in Report No. 951108-001, *Canister Storage Building Construction Restart Readiness Assessment* (Ares 1996), which contains a requirement to document the acceptability of the existing basemat in Appendix B (Item No. 2.12, Page B-3). Before the resumption of CSB construction, detailed inspections and evaluations of the acceptability of the existing basemat foundation will be performed and documented.

The architect-engineer has prepared a statement of work document and will participate in the walkdown. This will ensure that the report will contain sufficient information for the architect-engineer to adequately evaluate the acceptability of the existing basemat. Depending on the extent of degradation of the existing structure, repairs may be necessary. Detailed instructions to ensure bonding to the existing structure will be provided on the architect-engineer's construction drawings and specifications.

2.5 PROCESS DESCRIPTION

2.5.1 Baseline Operations

Upon arrival at the CSB from the Cold Vacuum Drying Facility, the cask containing the MCO is unloaded into a service pit by the gantry crane where the MCO is reinerted and prepared for vented storage. Reinerting consists of removal of all gases in the MCO and refilling the MCO space with an inerting gas. A service pit portable ventilation enclosure provides confinement during these operations. Once reinerting operations are completed, the MCO is prepared for vented staging and placed in a storage tube by the MHM. The MHM contains an on-board confinement system to mitigate against the effects of an accidental drop of an MCO during placement in a storage tube. The storage tube is inerted to a slightly positive pressure during staging of the MCOs. MCOs are sealed at the Hot Vacuum Conditioning Facility before being placed in interim storage. All storage tubes contain impact limiters to mitigate the effects of an accidental drop of an MCO.

2.5.2 Unresolved Process Issues

It is possible that if a subset of MCOs are certified to contain only undamaged fuel, then the hot vacuum conditioning step could be bypassed and those MCOs could be placed directly into sealed interim storage. Sealed MCO storage eliminates the reinerting operations at the loading pit and the surveillance and monitoring steps associated with inerted storage tubes. Confinement control is also facilitated. For such MCOs, use of the loading pit enclosure and its confinement function during reinerting operations may be eliminated if the superstructure has been hardened.

2.5.3 Process Impact on Below-grade Canister Storage Building Design

The below-grade portion of the storage vault design will not be affected by the resolution of these process issues.

2.6 CONFINEMENT SYSTEMS

2.6.1 Confinement System Descriptions

Because the fuel cladding is not intact, the primary confinement feature for the SNF at the CSB consists of the stainless-steel MCOs. Initially, the MCOs to be stored in the CSB will have undergone cold vacuum drying. The MCOs processed by cold vacuum drying are assumed to contain a slight amount of moisture that will contribute to some radiolytic and corrosion-induced hydrogen gas generation. The MCOs will be allowed to vent through high-efficiency particulate air (HEPA) filter plugs located atop the MCO closure head during their staging period in the storage tubes.

The secondary confinement system consists of the storage tube into which the MCOs are vented. The storage tubes are inerted and manually vented to the operating area through testable HEPA filters located on special service carts that periodically check and recharge the inerting gas in the storage tubes. If the pressure inside of a tube rises above 5 psig as a result of hydrogen generation, the tubes will vent to the operating area through a relief valve and HEPA filter in the floor plugs. The storage tube and the concrete plug above the tube are safety class 1 and have been designed to remain intact and functional when subjected to a DBE.

2.6.2 Confinement During Multicanister Overpack Handling Operations

During MCO reinerting operations, air will flow from the operating area through the portable enclosure HEPA filtration system. Confinement is provided by the portable enclosure ventilation system from the time the MCO cask is opened until the MHM is brought into the loading pit to remove the MCO to its storage tube. During the brief time that the portable enclosure is moved out of the way of the MHM, the reinerted vented MCO will be covered by a shield valve that will continue to provide a confinement barrier.

During MCO handling by the MHM, air will flow from the operating area or from the storage tube into the MHM cask space and will be exhausted through the MHM on-board HEPA filtration system. Confinement in the storage tube is maintained by the MHM ventilation system during MCO handling.

2.6.3 Conclusions

Neither of these design approaches to confinement problems should impact the below-grade design of the CSB regardless of their final design. As discussed further in Section 3.4, "Accident Analysis," all accidents require safety class 1 storage tubes and a support system design to preclude a common-cause failure of multiple MCOs. The base slab and walls that provide support to the deck and to the MCO storage tubes are designed to safety class 1 criteria.

2.7 SAFETY SUPPORT SYSTEMS

This section identifies and describes the principal system that performs safety support functions (i.e., those safety support functions that are not part of specific processes).

2.7.1 Safety Support System Description

The safety support functions of the CSB are provided by backup electrical power and fire protection and monitoring systems. The design of these systems is not included in the below-grade vault package. A fire hazards analysis for the below-grade portion of the structure indicates that no fire prevention or mitigative features are required (WHC 1996).

2.10 REFERENCES

2.10.1 Documents

- Ares, 1996, *Canister Storage Building Construction Restart Readiness Assessment*, Report No. 951108-001, Ares Corporation, Richland, Washington.
- WHC, 1992, *Hanford Waste Vitrification Plant Preliminary Safety Analysis Report*, WHC-SD-HWV-PSAR-001, Rev. 1, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1994, *Hanford Waste Vitrification Plant Canister Storage Building Preliminary Safety Analysis Report Addendum*, WHC-SD-HWV-PSE-001, Rev. 0A, --Westinghouse Hanford Company, Richland, Washington.
- WHC, 1995a, *Spent Nuclear Fuel Canister Storage Building Conceptual Design Report*, WHC-SD-W379-CDR-001, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1995b, *Performance Specification for the Spent Nuclear Fuel Canister Storage Building*, WHC-S-0425, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1995c, *Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements*, WHC-SD-SNF-DB-003, Rev. 1, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1996, *Fire Hazard Analysis for the Canister Storage Building Storage Vaults*, WHC-SD-SNF-FHA-002, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

2.10.2 Drawings

- H-2-117795, 1993, *Architectural/Plumbing Canister Storage Building Operating Area Plan*, Rev. 1, Fluor Daniel, Incorporated, Richland, Washington.
- H-2-117798, 1993, *Architectural Canister Storage Building, Building Sections*, Rev. 1, Fluor Daniel, Incorporated, Richland, Washington.

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3.0 HAZARD AND ACCIDENT ANALYSES

3.1 INTRODUCTION AND SUMMARY

This section summarizes the analyses performed to support the CSB conceptual design. The methods used to determine hazard category for the facility, and select accident scenarios for analysis, are described. The analyses of the selected accidents are then summarized, with emphasis on the effect of the accident on below-grade structure design.

The hazard category of the facility was determined to be hazard category 2. A preliminary hazards analysis (PHA) was performed and the resulting credible accident scenarios were sorted into groups based on inventory at risk. A representative accident from each group was chosen for analysis. The groups were

- Overpressure in an MCO
- MCO failure during handling (one MCO)
- MCO failure during handling (two MCOs)
- Loss of MCO confinement during staging or storage
- Accumulation of hydrogen gas
- Multiple MCO-tube failure

The MCO overpressure accident, and the MCO handling accidents, if unmitigated by safety barriers, resulted in unacceptable radiological dose consequences to the 100-m receptor. A safety class ventilation system, exhausted to a stack, was deemed to be sufficient to reduce the doses to acceptable levels.

The events involving hydrogen gas accumulation and multiple MCO-tube failures caused by external events were not analyzed in detail. However, it was concluded that unmitigated releases from these events would exceed those for the accidents analyzed, and therefore safety class preventive or mitigative features would be necessary.

The CSB design includes safety class 1 designations for the MCO, the storage tube, the operating floor, and the vault floor and walls. The required SSCs for mitigating releases from, or preventing, the accidents considered affect only systems and structures in the above-grade portion of the facility. The safety class designation of the below-grade features is unaffected.

3.2 REQUIREMENTS

This section lists the design codes, standards, regulations, and DOE orders that are required for establishing the safety basis for the CSB. The intent is to provide only the requirements that are specific for Chapter 3.0 and that pertain to the hazard and accident analysis.

3.2.1 U.S. Department of Energy Orders and Standards

The following DOE orders and standards are applicable to the safety analysis.

- DOE Order 5480.23, 1992, *Nuclear Safety Analysis Reports*. Attachment 1 of this Order establishes the format and content for Chapter 3.0, the hazard and accident analysis chapter, of a safety analysis report. The attachment also describes the methodology to be used in the performance of accident analyses, including guidance on hazards identification and assessment, the spectrum of accidents to be considered, and the performance of the consequence analyses.
- DOE Order 6430.1A, 1989, *General Design Criteria*. This Order provides requirements for the identification of "safety class items," which, by Westinghouse Hanford Company (WHC) procedures, are equivalent to safety class 1 SSCs.
- DOE-STD-1027-92, 1992, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*. This standard establishes guidance for the preparation and review of hazard categorization and accident analyses techniques as required by DOE Order 5480.23. Of particular importance to Chapter 3.0 of the CSB safety analysis report is the guidance relative to the hazard categorization methodology and the accident analyses techniques appropriate to the graded approach required by the Order.
- DOE-STD-3009-94, 1994, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*. This standard supplements DOE Order 5480.23 by providing guidance specific to nonreactor nuclear facilities. In this regard, the standard provides more detailed information on the performance of accident analyses for Hazard Category 2 and 3 facilities.

3.2.2 U.S. Nuclear Regulatory Commission Rules and Guidance

The following NRC rules and guidance are applicable to the safety analysis.

- 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," *Code of Federal Regulations*. This rule is used for the licensing of ISFSIs. Section 72.106, "Controlled Area of an ISFSI or MRS," establishes a limit of 5 rem at the controlled area boundary for any design basis accident. Section 72.122, "Overall Requirements," requires that the design bases for SSCs important to safety must reflect appropriate combinations of the effects of normal and accident conditions, and the effects of natural phenomena. Section 72.24, "Content of Application: Technical Information," provides requirements in 72.24(m) for the analyses of accidents and natural phenomena events that may result in a dose at the controlled area boundary.

- NRC Regulatory Guide 3.48, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)*. This guide establishes the format and content for materials license applications for 10 CFR 72 facilities, including the requirements for Chapter 8, "Accident Analyses." For this chapter, Regulatory Guide 3.48 states that ANSI/ANS-57.9-1984, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)*, defines four categories of design events that provide a means of establishing design requirements to satisfy operational and safety criteria. This standard was reissued in 1992 and retitled *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)* (ANSI/ANS 1992).

3.2.3 Industry Consensus Standards

The following standard is applicable to the safety analysis.

- ANSI/ANS-57.9-1992, 1992, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)*. Section 2 of this standard, "Definitions," defines Design Event categories that are to be used as described in Section 5, "ISFSI Performance Requirements," to evaluate the radiological protection provided by the facility. Design Events I and II cover normal and off-normal events. These events are usually addressed by summarizing analyses that confirm the SSCs are appropriately designed to accommodate the events without challenge to the design. Design Events III and IV cover a range of postulated accident events for which radiological calculations are performed to confirm that onsite and offsite doses are within acceptable limits as defined in 10 CFR 72.

3.3 HAZARD ANALYSIS

This section describes the hazard identification and evaluation performed for the facility. It is a summary of the information provided in WHC-SD-SNF-PSE-002, *Spent Nuclear Fuel Project Canister Storage Building Preliminary Safety Evaluation* (WHC 1995).

3.3.1 Hazard Category

The CSB has been assigned a hazard category of 2 in accordance with DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*. The final hazards category for the CSB was derived in WHC-SD-SNF-HC-007, *Hazard Category Analysis for the Canister Storage Building* (Kummerer 1995). That document was prepared in accordance with WHC-CM-4-46, *Nonreactor Facility Safety Analysis Manual*, Section 4.0, "Determining and Documenting Facility Hazard Category," the WHC guide for implementing and documenting the requirements of DOE Order 5480.23 and DOE-STD-1027-92.

The hazard category was determined by identifying the facility radioactive materials inventory at risk for release, unmitigated by any safety features. Currently, the inventory to be stored in the CSB is limited to the total fuel stored in the K East and K West Basins (approximately 2,100 t of fuel).

The inventory at risk for release was taken to be a bounding estimate of the particulate material remaining in the MCOs during staging and storage. The fuel elements and large fuel pieces were excluded from the inventory at risk for release because airborne suspension from the fuel surfaces would not occur. Particulate material will be washed from the fuel before packaging at K Basins. The vacuum drying process will remove all but a nominal amount of free water. This will minimize the formation of additional particulate material during transportation, handling, and staging.

The total inventory at risk for release was 2.56×10^6 g. It was assumed that the particulate material at risk for release had the same radionuclide content as the fuel. Using these assumptions, the sum of ratios to the Hazard Category 2 threshold quantities of DOE-STD-1027-92 was 19.0. If the sum of the ratios exceeds 1.0, the facility is designated Hazard Category 2.

3.3.2 Preliminary Hazards Analysis

A PHA was conducted by an assessment team to identify potential accidents in accordance with WHC-CM-4-46, Section 5.0, "Preliminary Safety Evaluation." The PHA satisfies the requirements of DOE Order 5480.23 and its implementing standard, DOE-STD-3009-94, for a systematic, rigorous technique for identifying the range of potential hazards posed by the operation of the facility. The PHA is documented as Appendix A of WHC-SD-SNF-PSE-002 (WHC 1995).

The events identified by the PHA were further evaluated by establishing a potential accident and sequence, potential targets or consequences, and mitigating features (engineered and administrative), and qualitatively estimating the probability of occurrence. The event evaluation provides a basis for establishing barriers for mitigating hazardous material releases to onsite and offsite receptors. The PHA establishes the highest radiological consequence level for each event.

The PHA process systematically examined the CSB and facility activities, and identified the hazards associated with postulated accidents. Criteria for selecting accidents for further study were based on the qualitative consequences and likelihoods. An alphanumeric system was used to designate the severity with the following "S" rankings characterizing safety consequences:

- S0 No effect outside the facility confinement systems and no safety concerns for the facility worker, the onsite worker, or members of the general public
- S1 Potential industrial injury, radiological dose consequences, or chemical exposure to the facility worker; limited environmental discharge of hazardous material outside the facility

- S2 Potential significant radiological dose consequences or chemical exposure to the onsite worker at the nearest adjacent facility; significant environmental discharge of hazardous material within the Hanford Site boundary
- S3 Potential significant radiological dose consequences or chemical exposure to the offsite population; environmental discharge of hazardous material outside the Hanford Site boundary or to the groundwater.

Only those items ranked as categories S2 and S3 were considered to be sufficiently significant to require consideration for additional analysis. Category S3 consequences may have the potential for significant radiological dose or chemical exposure to the offsite population, or for environmental discharge of hazardous material to the Hanford Site boundary or to the groundwater. Category S2 consequences may have the potential for significant radiological dose or chemical exposure to the onsite worker at the nearest adjacent facility, or for significant environmental discharge of hazardous material within the Hanford Site boundary.

Accidents in categories S2 and S3 were combined into major groupings based on the inventory at risk. The seven accident types are as follows:

- Overpressure in an MCO
 - MCO pressurization
 - Pressure/contamination release during unbolting of cask lid
 - Damage to MCO from dropped cask lid
 - Pressure/contamination release during unbolting of flanges
 - MCO pressurization with purge gas
- MCO failure during handling (one MCO)
 - Runaway transport vehicle
 - Cask/MCO tip over or drop from truck
 - Cask/MCO drop into service pit
 - MCO drop onto shipping cask without impact limiter
 - Misalignment of MCO during lowering or raising process
 - Closure of floor valve or MHM closure valve at wrong time (MCO in intermediate position)
- MCO failure during handling (two MCOs)
 - MCO drop on operating area floor
 - Placement of MCO in wrong tube
 - MCO drop into storage tube
- Loss of MCO confinement during staging or storage
 - MCO structural failure or rupture disk relief during staging
 - MCO structural failure during storage

- Accumulation of hydrogen gas
 - Hydrogen burn in MCO during purging or in purging equipment
 - Fire in MHM/tube caused by hydrogen venting from MCO
 - Hydrogen burn in MCO
 - Hydrogen burn in storage tube during staging
 - Hydrogen burn in MCO
 - Hydrogen burn in storage tube during storage
- Multiple MCO/tube failure
 - Structural failure of the CSB and/or SSCs within the CSB.

3.4 ACCIDENT ANALYSIS

This section presents the formal processes used in development of the potential accidents identified in Section 3.3, "Hazards Analysis."

3.4.1 Methodology

The radiological consequences of the accidents, unmitigated by any safety features, were determined and compared to the criteria listed in Table 3.4-1.

Table 3.4-1. Safety Classification Criteria.

Effective dose equivalent
Offsite guidelines: 0.005 Sv (0.5 rem)
Onsite guidelines: 0.05 Sv (5.0 rem)

Note: This information is from WHC-CM-4-46, Nonreactor Facility Safety Analysis Manual, Section 9.0, "Safety Classification of Structures, Systems, and Components," Westinghouse Hanford Company, Richland, Washington.

These criteria for identifying safety class SSCs were developed by WHC to implement the guidance of DOE Order 6430.1A, Section 1300-1.4, "Guidance on Limiting Exposure of the Public." They are consistent with requirements of 10 CFR 72 in establishing a limit of 5 rem at the controlled area boundary. The acceptance guidelines are given in terms of whole body, 50-year committed effective dose equivalent for the onsite and offsite receptors. The onsite receptor is defined as being 100 m from the point of release in the direction of highest dose. The offsite receptor is at the Hanford Site boundary in the worst-case direction.

If the radiological dose for the unmitigated case exceeded the guideline, mitigating safety features, with appropriate safety class designation, were suggested. The dose consequences were recalculated taking appropriate credit

for the mitigating safety features to verify that the mitigated doses would satisfy the guidelines.

For each accident scenario, the airborne radiological dose calculated using the methods described was compared to the appropriate acceptance guideline limit. For this comparison, the onsite and offsite limits of 5.0×10^{-2} Sv (5.0 rem) and 5.0×10^{-3} Sv (0.5 rem), respectively, were used as the acceptance guidelines for all events. Event frequencies were not considered at this stage in the CSB design because sufficient data are not available to establish appropriate frequencies.

None of the accidents identified by the PHA presented a hazard of radiological releases to the ground or groundwater. The MCO will be drained and the fuel dried to remove excess water before shipping to the CSB. Therefore, the potential for liquid releases to the ground during receiving, handling, staging, or storage does not exist.

Based on the CSB design and operating information currently available, no use of toxic chemicals has been identified. The risk of toxic chemical exposure will need to be considered as more detailed information on each phase of the process is developed.

3.4.2 Design Basis Accidents

This section discusses the representative accidents analyzed from each of the major groupings, and accidents initiated by human-caused external events or natural phenomena. The accidents are discussed relative to their effect on the design basis of the below-grade structure.

3.4.2.1 Representative Accidents Analyzed. The details of the analyzed accidents are provided in WHC-SD-SNF-PSE-002 (WHC 1995, Section 4). They are summarized in the following sections.

The following assumptions were common to all accidents.

- The MCO contains 140 MKIV fuel elements (3,336 kg fuel).
- Five percent of the fuel mass, 166.8 kg, is in particulate form.
- The 50-year committed effective dose equivalent is 3.1×10^3 Sv/g fuel. This is the value calculated for K East Basin MKIV canister, the fuel category with the highest specific dose.
- For ground-level releases, the onsite receptor was located 100 m east of the facility. For releases mitigated by elevation of the release point, the maximum dose was found to be at 230 m west of the facility. Therefore, the mitigated doses reported for the onsite receptor are those calculated for that distance.
- The maximum offsite receptor was 18.3 km east of the facility.

Overpressure in a Multicanister Overpack. This accident postulated that inadvertent addition of purge gas at too high pressure, or production of excess chemical gases inside the MCO, produces internal pressure exceeding the capacity of the rupture disk. The disk ruptures and MCO gases carrying fuel particulate are expelled into the CSB.

To develop the source term, the assumed particulate loading of the MCO was treated as a pressurized powder, with an airborne release fraction (ARF) of 1.73×10^{-3} and a respirable fraction (RF) of 0.35. The amount of material released was 0.1 kg.

When the release was unmitigated by any safety barriers, the radiological dose consequence to the onsite receptor was 3.5 Sv (350 rem). Therefore, the dose exceeds the 5.0 rem criterion at 100 m and must be mitigated. When credit was taken for a building ventilation system, exhausted to a stack with effective release height of 36.6 m (120 ft), the dose consequences to the onsite receptor were reduced to 0.43 rem.

Dose consequences at the Site boundary for the unmitigated release were 0.14 rem. This is less than the 0.5 rem guideline for offsite dose consequences required by WHC guidelines.

Effect on Below-grade Structure. A decision to provide a safety class 2 superstructure, ventilation system, and stack, or other confinement closer to the source of release, will not affect the design of the below-grade structure.

Multicanister Overpack Failure During Handling (One Multicanister Overpack). The scenario chosen to represent this group of accidents was dropping an MCO from the crane to the operating area floor. The MCO was assumed to fail, exposing all of the fuel and particulate inventory to the environment. No credit was taken for partial containment of the release by the damaged MCO.

Airborne suspension of particulate material was assumed to arise from two mechanisms: 1) a quantity of particles lofted by the energy of the impact, and 2) an additional quantity suspended over time by airflow over the spilled MCO contents. The $ARF \times RF$ for the immediate release, derived using an empirical expression, was 2.42×10^{-4} .

The aerodynamic entrainment factor for the suspension over time was $4 \times 10^{-5}/h$. For the onsite receptor, the release was assumed to continue over an 8-hour period, the length of one shift. For the receptor at the Site boundary, the hazard was assumed to continue for 24 hours. It is assumed that within that time, emergency measures would have been taken to prevent human occupation of the area affected by the release.

When the release was unmitigated by any safety barriers, the radiological dose consequence to the onsite receptor was 2.0 Sv (200 rem). Therefore, the dose exceeds the 5.0 rem criterion at 100 m and must be mitigated. When credit was taken for a building ventilation system, exhausted to a stack with effective release height of 36.6 m (120 ft), the dose consequences to the onsite receptor were reduced to 0.23 rem.

Dose consequences at the Site boundary for the unmitigated release were 0.24 rem. This is less than the 0.5 rem guideline for offsite dose consequences required by WHC guidelines.

Effect on Below-grade Structure. A decision to provide a safety class 2 superstructure, ventilation system, and stack, or other confinement closer to the source of release, will not affect the design of the below-grade structure. The walls and floor that provide support to the storage tubes and MCOs are designed to be safety class 1 to protect the containment function of the tubes and MCOs. The safety class 1 design features will assure that the operating deck floor can sustain the impact of a dropped MCO, or safety class 1 impact limiters will be used.

Multicanister Overpack Failure during Handling (Two Multicanister Overpacks). MCO failure accidents involving two MCOs were characterized by a drop of an MCO onto another MCO already in place in a storage tube. As in the accident involving only one MCO, it was assumed that the entire particulate inventory of both MCOs was available for airborne suspension. No credit was taken for holdup by the tube.

The release quantity, source term and radiological dose consequences were derived using the same assumptions as for the single MCO failure accident. Therefore, the dose consequences for the various receptors and mitigated case were twice those from that accident.

When the release was unmitigated by any safety barriers, the radiological dose consequence to the onsite receptor was 4.1 Sv (410 rem). Therefore, the dose exceeds the 5.0 rem criterion at 100 m and must be mitigated. When credit was taken for a building ventilation system, exhausted to a stack with effective release height of 36.6 m (120 ft), the dose consequences to the onsite receptor were reduced to 0.46 rem.

Dose consequences at the Site boundary for the unmitigated release were 0.47 rem. This is less than the 0.5 rem guideline for offsite dose consequences required by WHC guidelines.

Effect on Below-grade Structure. A decision to provide a safety class 2 superstructure, ventilation system, and stack, or other confinement closer to the source of release, will not affect the design of the below-grade structure. The walls and floor that provide support to the storage tubes and MCOs are designed to be safety class 1 to protect the containment function of the tubes and MCOs. The safety class 1 design features will assure that the operating deck floor, the tube, and the lower deck floor can sustain the impact of dropping an MCO into a tube, or safety class 1 impact limiters will be used.

Accumulation of Hydrogen Gas. It is postulated that, because of insufficient water removal during vacuum drying, incomplete conditioning, and/or delays in transport and handling, chemically and radiolytically generated hydrogen gas could build up in the MCO. While the MCO is sealed, and inerted, the potential for a hydrogen burn does not exist. However, when the MCO is opened for servicing, or if MCO seals fail, the escaping gases mixing with air could allow a flammable mixture to exist.

Detailed analysis of a hydrogen burn in the CSB was not performed at this stage of design, pending further design details and knowledge of fuel condition. However, it is acknowledged that a significant portion of the inventory of an MCO would be at risk for release for this event.

Effect on Below-grade Structure. It is expected that analysis of this event would lead to at least Safety Class 2, and probably Safety Class 1 measures to prevent development of a flammable gas mixture and its subsequent ignition. These safety class designations would be given to SSCs in the receiving and operating areas of the CSB (above grade). Therefore, the consequences of the hydrogen accumulation scenarios do not affect the design of the below-grade portion of the CSB.

Loss of Multicanister Overpack Confinement during Staging or Storage. This accident scenario postulated a breach in an MCO in its tube during staging or storage, allowing air entry into the MCO. The mechanism causing the breach was unspecified, but the opening was assumed to remain small, about the area of a 2.5-cm- (1-in.-) diameter circular hole. It was also assumed that conditioning was imperfect, and that significant fuel surface would be available for oxidation.

The mechanism for release of MCO gases to the facility was taken to be the normal atmospheric breathing rate for the Hanford Site, 0.0046 of the volume of the breathing vessel. This was based on the design assumption that the tube would be vented to the building. The breached MCO would release its gases, and entrained particulates, into the tube, and the tube would exchange gases with the building air.

The MCO gases were assumed to contain 10 mg/m³ fuel particulate. This is a typical upper value for suspension of dusts in still air. The release rate was 0.037 mg/day. Radiological dose consequences from this release over 1 day were calculated to be 4.3×10^{-5} rem at 100 m. Consequences at the Site boundary were not calculated because they would be less than the onsite doses.

Effect on Below-grade Structure. The consequences of this event were acceptable without any mitigating features, so no safety class SSCs were called for. Moreover, changes have been made in the design of the MCO/tube since this scenario was analyzed. These changes further limit the potential for releases from a breached MCO.

During staging, the MCO will be open to the tube through a filtered port, and the tube will be sealed. Both MCO and tube will have an inert gas cover. During storage, the MCO will be inerted and sealed. The tube will contain air. In the event of an MCO breach during storage, the tube will provide containment of gas and particulate releases from the MCO and prevent their release into the environment. Therefore, an MCO breach during staging or storage does not affect the design of the below-grade structure.

3.4.2.2 Externally Initiated and Natural Phenomena Accidents. These accidents were not analyzed in the preliminary safety evaluation but are discussed here to assess their effects on the below-grade design.

Multiple Multicanister Overpack-Tube Failure. The scenario is a postulated failure and collapse of the building superstructure from external human-caused or natural phenomena. It is assumed that the collapse could cause a common mode failure of multiple tubes and MCOs.

Effect on Below-grade Structure. The walls and floor that provide support to the storage tubes and MCOs are designed to be safety class 1, to protect the containment function of the tubes and MCOs. The safety class 1 design will include assurance that the tubes and MCOs maintain their integrity after superstructure collapse.

Failure of the Multicanister Overpack Handling Machine, Ventilation Hood, or Building Superstructure from a Tornado Missile. A transport cask containing an MCO will be placed in the service pit when it arrives at the CSB. The transport cask will be opened and the flanges removed from the filter, rupture disk, and connection ports in preparation for purging and moving the MCO to the staging tube. A ventilation hood over the service pit will contain releases during these operations. After servicing, the MCO will be moved to its staging tube by the MHM.

Containment of releases from accident scenarios presented previously in this section mandates that either the building superstructure or the MHM and ventilation hood be designated safety class 1. This would include a design hardened against damage from the design basis tornado.

Effect on Below-grade Structure. NPH wind loads and missiles associated with a tornado will be accommodated by the design. Safeguarding against tornado-driven missiles will require hardening the local confinement barriers at the MCO service pit and on the MHM or hardening the superstructure. Alternately, using a probabilistic risk assessment approach may reduce the requirement for tornado-resistant SSCs. Hardening of the superstructure could increase the loading on the below-grade walls and foundation. The impact associated with increased loading to the below-grade structure will be evaluated as part of the decision process. The decision on the design approach will be made during definitive design.

Loss of Convective Cooling to the Vault. The CSB storage vault will be designed such that natural air circulation will maintain MCO temperatures low enough to prevent failure. Unacceptably high temperatures could cause gas expansion and accelerated gas generation from chemical reactions that could result in pressurization, and subsequent breach, of the sealed MCOs.

If debris from high winds, or snow and ashfall accumulations, plugged the inlet and/or outlet to the vault, the natural airflow would be hampered or completely stopped. The MCOs would heat up and could reach a temperature at which chemical reactions of the fuel are accelerated. The product gases of the reactions, as well as the increase in temperature, would cause increased internal pressure in the vessel. There is a potential for one or more MCOs to breach, releasing MCO gases to the storage tube that could mix with the air in the sealed tube. Ultimately, the tube could also pressurize and vent through the relief valve in the shield plug, releasing the gases to the operating deck.

During staging, the MCOs will be vented to their respective storage tubes, and both MCO and tube will be filled with inert gas. The MCOs will contain bound water that would be released at elevated temperatures, as well as residual free water that would vaporize. The water reacting with the fuel would increase temperatures further. The MCO is vented, so MCO overpressure resulting in MCO breach would not occur. However, the gases would pressurize the tube and the tube would vent through the relief valve in the shield plug.

During storage, the fuel in the MCO will be covered with an inert gas and the MCO will be sealed; the storage tube will contain air. The hot conditioning process will have removed both bound and free water and covered free metal surfaces with a nonreactive oxide layer. Therefore, the potential for accelerating chemical reactions is less during the storage phase.

The additional hazard from extended loss of cooling is the heatup of the vault walls to a temperature that would compromise their structural strength.

Effect on Below-grade Structure. The below-grade structure was originally designed for the heat loads expected from the vitrified high-level waste. The heat loading density for one vault containing SNF MCOs is less than it would have been with the high-level waste canisters. Wall temperatures reached during a postulated failure of convective cooling, therefore, would not exceed those considered for the same accident for the vitrified waste mission.

Future analysis will determine the heatup rates and potential maximum temperature of the fuel in the event of loss of convective cooling. If the analysis reveals that temperatures high enough to lead to MCO and/or tube breach are credible, the response time to restore convective cooling, and thereby preclude hazardous fuel temperatures, will be characterized.

3.4.3 Beyond Design Basis Accidents

Beyond design basis accidents for the below-grade portion of the CSB would include those events that exceed the severity of the NPHs described in Section 1.0. Beyond design basis accidents normally are evaluated in a safety analysis report to assess the residual risk of the facility. To date, no beyond design basis accidents have been evaluated for the CSB. The evaluations will be completed when definitive design for the facility is available and will be included in the final safety analysis report.

3.4.4 Safety Class Design Features

The Preliminary Safety Evaluation (WHC 1995) identifies safety class 2 ventilation with a stack (and without high-efficiency particulate air filters) as a mitigating feature. This feature would provide the required mitigation for all accidents analyzed. Alternative measures associated with each accident that would provide the required level of protection are as follows.

For the MCO drop accidents:

- Design the MCOs as safety class 2 barriers; the design features may include impact limiters, and possibly an air or water cushioning concept, to prevent MCO failure upon a drop
- Provide safety class 2 confinement barriers during MCO handling such that radionuclides that might be released upon an MCO drop and failure are not released to the CSB operating area (e.g., during the lowering of an MCO into a storage tube, this confinement could be provided by the storage tube and the MHM with interfacing seals).

For the MCO overpressure events:

- Design the MCO as a safety class 2 barrier to maintain confinement for the maximum overpressure event
- Provide a safety class 2 confinement barrier for all areas where the MCO is handled outside the transfer cask
- Use a fuel vacuum drying process prior to staging and also possibly use getters, recombiners, or hydrogen-permeable materials such that the risk of this event can be considered not credible.

The selection of the design approach for mitigation of the accidents will be performed during further development of the final design. The design approach will consider the operating philosophy and cost-effectiveness of the design alternatives and will be coordinated with the accident analysis.

In addition, all accidents require safety class 1 storage tubes and support structures to preclude accidents propagating to adjacent MCOs. Safety class 1 assignment must also be made to those SSCs necessary to prevent loss of double-contingency protection against criticality.

The impact of the safety class 1 and 2 SSCs (identified above) on the below-grade construction is that the walls and floor that provide support to the MCO storage tubes and MCOs need to be safety class 1. This is already a requirement for these structures. This is also the highest safety classification category that can be applied to this design and construction. The only remaining issue is whether the specific design features are adequate to accommodate the required safety class 1 and 2 SSCs.

While there are decisions and issues that remain to be resolved relative to accident prevention and mitigation features of the CSB, these issues can and will be resolved such that the below-grade design of the CSB is not affected. The bases for this conclusion are as follows.

- A decision to provide either a safety class 2 CSB, ventilation system, and stack or closer-in confinement will not impact the design of the below-grade CSB. The first option will require a building of similar structural design but of less infiltration leakage. Also, an air-handling unit and a stack would be required. The stack may be a significant structure, but it will not be

physically tied to the CSB structure. If it is decided to locate the confinement barriers closer to the release points, the impact will be to systems and components within the CSB.

- The open issues relative to accident analyses and SSC safety classification involve implementation of "important to safety," potential new accidents, and refinement of existing accidents, some of which could lead to increased severity. The probable maximum impact of the resolution of these issues would require that the safety classification assignment for some SSCs would be increased from safety class 2 to safety class 1.
- The requirements of DOE Order 6430.1A for "safety class items" would apply to those SSCs classified as safety class 1 because they are important to safety. The natural phenomena design requirements of DOE Order 5480.28 also would increase from those for Performance Category 2 to those for Performance Category 3.

In summary, the CSB Preliminary Safety Evaluation provides adequate information and conclusions to support construction completion of the below-grade portion of the CSB because

The identified below-grade portion of the CSB already has been designated to the highest level safety classification, i.e., safety class 1

While open issues relative to accident analyses and design options for accident prevention and mitigation remain to be resolved, they can and will be resolved without affecting the design details of the identified below-grade portion of the CSB.

3.5 REFERENCES

- DOE Order 5480.28, *Natural Phenomena Hazards Mitigation*, U.S. Department of Energy, Washington, D.C.
- Kummerer, M., 1995, *Hazard Category Analysis for the Canister Storage Building*, WHC-SD-SNF-HC-007 Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1995, *Spent Nuclear Fuel Project Canister Storage Building Preliminary Safety Evaluation*, WHC-SD-SNF-PSE-002, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- WHC-CM-4-46, *Nonreactor Facility Safety Analysis Manual*, Westinghouse Hanford Company, Richland, Washington.

4.0 SAFETY STRUCTURES, SYSTEMS, AND COMPONENTS

4.2 REQUIREMENTS

This section lists the design codes, standards, regulations, and DOE orders that are required for establishing the safety basis of the facility. The intent is to provide only the requirements that are specific for this chapter and pertinent to the safety analysis.

4.2.1 U.S. Department of Energy Orders and Standards

The following DOE orders and standards are applicable to the safety analysis.

- DOE Order 6430.1A, 1989, *General Design Criteria*. Division 1300-3, "Safety Class Criteria," provides design, fabrication, and testing standards for safety class items.
- DOE Order 5480.28, 1993, *Natural Phenomena Hazards Mitigation*. This order establishes mitigation requirements for NPHs and target probabilistic performance goals based on the facility performance category. Additional discussion of the NPH performance criteria is provided in Section 1.2.2.
- DOE-RL-HPS-SDC-4.1, 1993, *Standard Architectural-Civil Design Criteria, Design Loads for Facilities*. This Hanford Site plant standard provides minimum requirements for the design of the below-grade portion of the CSB. As stated in WHC-SD-HWV-PSE-001, *Hanford Waste Vitrification Plant Canister Storage Building Preliminary Safety Analysis Report Addendum* (WHC 1994), this is one of the main reference standards for facility design. Section 3.1 and Section 3.3 of this standard provide design criteria loads and load combinations for safety class 1 SSCs. These include nominal dead, live snow, and soil loads; normal operating loads; and natural phenomena loading of extreme wind, earthquake, ashfall, and flood. Natural phenomena loads listed on Table 1.5-1 of Section 1.5, "Natural Phenomena Threats," are higher than SDC-4.1 loads, particularly those associated with tornado wind and PMP effects. For example, the tornado wind load from Table 1.5-1 is given as 322 km/h (200 mi/h) total, versus the SDC-4.1 wind load of 145 km/h (90 mi/h). This higher wind load on the superstructure will transmit higher loads to the below-grade portion of the CSB.

4.2.2 U.S. Nuclear Regulatory Commission Rules and Guidance

The following NRC rule and guidance is applicable to the safety analysis.

- 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," *Code of Federal Regulations*. The design bases for items identified as important to safety must reflect appropriate combinations of the

effects of normal and accident conditions and the effects of natural phenomena. This is implemented by the application of the design criteria for safety class items in DOE Order 6430.1A to those items identified as important to safety. Additional discussion of the NRC criteria for mitigation of natural phenomena is provided in Section 1.2.3.

4.2.3 Industry Consensus Standards and Other Documentation

The following industry standard is applicable to the safety analysis.

- ANSI/ANS-57.9-1992, 1992, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)*. The standard includes requirements for the design of major buildings and structures including physical security and safety features for ISFSIs. A review of the performance and design requirements has been conducted and has concluded that the current CSB SNF design appears to be in general compliance with the exception of transportation package washdown, decontamination facilities, and safety class electrical power. Section 6.17.1.1 of this standard also requires that the most adverse dead load loading condition be increased by 5% from the estimated value. An increase in the dead load loading condition may impact the below-grade portion of the vault package. The current CSB NSF design does meet the requirements of ANSI/ANS-57.9-1992 in these areas.

4.2.4 Westinghouse Hanford Company Requirements

The following WHC standard is applicable to the safety analysis.

- WHC-CM-4-46, *Safety Analysis Manual*. Section 9 identifies the methodology for determining the safety classification of SSCs and the applicable design criteria. The safety class 1 requirements in Section 9 of WHC-CM-4-46 implement the criteria established in DOE Order 6430.1A for safety class items.

4.3 SAFETY CLASS STRUCTURES, SYSTEMS, AND COMPONENTS

The safety class SSCs for below-grade structures are described in this section to provide an understanding of the safety function and the suitability of the safety analysis inputs and assumptions. These safety class SSCs mitigate the potential consequence of the accidents summarized in Chapter 3.0.

The safety class SSCs are summarized in Table 4.3-1 (these items are designated as safety class 1 in accordance with WHC-CM-4-46). The remaining subsections provide details that expand upon the summary presentation in the table.

Table 4.3-1. Safety Class 1 Structures, Systems, and Components for the Below-grade Portion of the Canister Storage Building.

Safety class SSC	Safety function	Functional requirements	Performance criteria (TSR applicability)
Subsurface structure (concrete walls and base slab)	Prevent common cause failure of multiple MCOs	Withstand applicable NPHs and DBAs to extent required to prevent multiple MCO failures	Not applicable
Air intake and exhaust plenums	Provide enclosure for convective cooling of MCOs	Prevent overheating and failure of multiple MCOs	Not applicable
V-type anchor bolts	Provide embeds for future installation of insulating concrete on the inside walls of vaults 2 and 3	Prevent overheating and maintain structural integrity of the concrete	Not applicable
Carbon steel base slab embeds	Provide anchor point for seismic restraints for tubes	Prevent failure of one or more tubes and failure of one or more MCOs	Not applicable

DBA = Design basis accident.
MCO = Multicanister overpack.
NPH = Natural phenomena hazard.
SSC = Structure, system, and component.
TSR = Technical safety requirement.

4.3.1 Canister Storage Building Subsurface Structure

4.3.1.1 Safety Function. The CSB will contain MCOs in the below-grade space described in Chapter 2.0 as CSB vault area 1. The CSB is designed to prevent failure of MCOs. Failure of a single MCO could result in consequences over the safety class 2 guidelines. The potential consequences of an unmitigated release from the inventory at risk in the CSB was evaluated in Section 5.3 of WHC-SD-SNF-HC-007, *Hazard Category Analysis for the Canister Storage Building* (Kummerer 1995). Failure of a significant number of the MCOs could result in offsite consequences greater than the safety class 1 guideline. The CSB substructure is designed to prevent failure of the MCOs during the design basis accidents (DBAs) and NPHs given in Chapter 1.0, Section 1.1.2. Safety analyses resulting from implementation of NRC equivalency could determine that failure of a single MCO requires safety class 1 mitigation if an MCO is considered important to safety as defined in 10 CFR 72, Section 3.

4.3.1.2 System Description. The concrete base slab, exterior and interior walls, and air intake and exhaust plenums are safety class 1 features of the CSB. These items provide protection from all DBAs for the 220 tubes containing MCOs in vault 1.

4.3.1.3 Functional Requirements. The concrete substructure shall withstand all DBAs to the extent required to assure that the MCOs and the storage tubes are not damaged. The DBAs applicable to the substructure are the natural phenomena established for safety class 1 SSCs discussed in Section 1.2.2.

4.3.1.4 System Evaluation. The CSB substructure is designed to withstand the DOE-designated natural phenomena specified in Section 1.2.2. Additional natural phenomena requirements are associated with NRC equivalency. Notably, tornado hazards and suspended ashfall are added and PMP amounts increase. Evaluation of the substructure effects of suspended ashfall, higher PMP, tornado wind loading, pressure changes, and the tornado missiles upon the SNF CSB will be completed during the definitive design phase.

Structural calculations show that the currently designed load to the walls and base slab from the deck and MHM is not significantly different in the SNF CSB from what was expected under the HWVP CSB design. The HWVP CSB's superstructure was non-safety class. Under the HWVP CSB mission, the building's superstructure was not required to be hardened to protect any interior equipment from the NRC-related NPHs or DBAs. Evaluation of the effects of NPHs (e.g., suspended ashfall, PMP, tornado wind, and tornado missile impacts) will be evaluated during the definitive design phase. For example, tornado missile impact evaluation could require hardening of safety class equipment in the operating area, or hardening of the superstructure itself. Current design assumes the superstructure, intake structure, and exhaust stack loading to the substructure is similar to HWVP CSB design. Details of structural calculations for the current design will be available when the vault package design is approved for construction.

4.3.1.5 Controls (Technical Safety Requirements). Not applicable for this system.

4.3.2 Air Intake and Exhaust Plenums

4.3.2.1 Safety Function. The CSB will contain MCOs in the below-grade space described in Chapter 2.0 as CSB vault area 1. The CSB is designed to prevent failure of MCOs. Failure of a single MCO could result in consequences over the safety class 2 guidelines. The potential consequences of an unmitigated release from the inventory at risk in the CSB was evaluated in Section 5.3 of WHC-SD-SNF-HC-007 (Kummerer 1995). Failure of a significant number of the MCOs could result in offsite consequences greater than the safety class 1 guideline. The MCOs must be protected against overheating that could result in reactions with the fuel or compromise the structural integrity of the MCO. The air intake and exhaust plenums are integral features of natural convection cooling. Safety analyses resulting from implementation of NRC equivalency could determine that failure of a single MCO requires safety class 1 mitigation if an MCO is considered important to safety as defined in 10 CFR 72, Section 3.

4.3.2.2 System Description. The below-grade concrete structure forms the major portion of the natural convection cooling design for the CSB. Design of the natural convective cooling features is based upon two-dimensional thermal analysis using two-dimensional modeling software. Detailed design calculations will be available when the vault package is released for construction. Additional three-dimensional modeling will be completed at a later date to determine the temperature profile of the storage vault to verify thermocouple locations.

4.3.2.3 Functional Requirements. The air intake and exhaust plenums shall withstand all DBAs including natural phenomena to the extent required to ensure that cooling is maintained to the MCOs. The DBAs applicable to the intake and exhaust plenums are the natural phenomena established for safety class 1 SSCs discussed in Section 1.2.2.

4.3.2.4 System Evaluation. The air inlet and exhaust plenums will be designed to withstand the DOE-designated natural phenomena specified in Section 1.2.2. The above-grade structures that rest on the below-grade intake and exhaust plenums are designed to prevent plugging caused by ice, snow, ashfall, and windborne debris. Additional natural phenomena requirements are associated with NRC equivalency. Notably, tornado hazards and suspended ashfall are added and PMP amounts increase. Evaluation of the substructure effects of greater ashfall, higher PMP, tornado wind loading, pressure changes, and the tornado missiles upon the SNF CSB be performed during the definitive design phase.

Structural calculations show that the currently designed load to the air inlet and exhaust plenums is not significantly different in the SNF CSB from what was expected under the HWVP CSB design. Current design assumes the intake structure, and exhaust stack loading to the substructure is similar to HWVP CSB design. Details of structural calculations for the current design will be available when the vault package design is approved for construction. Tornado missile impact evaluation could require hardening of the air inlet structure or the stack for vault 1.

4.3.2.5 Controls (Technical Safety Requirements). Not applicable for this system.

4.3.3 V-Type Anchor Bolts

4.3.3.1 Safety Function. The structural concrete of the CSB substructure must be protected from temperatures in excess of 66 °C (150 °F) to prevent structural degradation. In the HWVP CSB design, this was accomplished through the addition of a layer of insulating concrete to the interior surfaces of the vault. For the SNF CSB, the insulating concrete layer is not required in vault 1 because MCOs stored therein present a lower thermal load. This situation allows elimination of the V-type anchor bolts from the interior surfaces of vault 1. However, in recognition of possible future missions for vaults 2 and 3, installation of carbon steel V-type anchor bolts in the interior surfaces of those vaults is under consideration.

4.3.3.2 System Description. Carbon steel V-type anchor bolt embeds, if required, will be installed in the interior surfaces of the walls in vaults 2 and 3 in a 20-cm (8-in.) staggered grid pattern, as shown on Drawing H-2-119278.

4.3.3.3 Functional Requirements. The carbon steel V-type anchor bolt embeds were designed for HWVP CSB use to withstand the DBE specified in Section 1.2.2. The seismic loadings for the anchors were designed based on expected loading for insulating concrete.

4.3.3.4 System Evaluation. The V-type anchor bolts are designed to withstand the DBE specified in Section 1.2.2.

4.3.3.5 Controls (Technical Safety Requirements). Not applicable for this system.

4.3.4 Carbon Steel Base Slab Embeds

4.3.4.1 Safety Function. The 220 tubes (and 5 overpack tubes) of vault 1 will be seismically restrained through the use of tube base assemblies that will be bolted to carbon steel embeds already in the base slab of the CSB once the tubes are installed in the deck. (The tube base assemblies are not part of the below-grade construction.) In vault 1 these embeds will serve the safety function of maintaining horizontal stability of the tubes containing MCOs during a DBE. The embeds are present in vaults 2 and 3 for future use. Their safety function for future use has not been evaluated.

4.3.4.2 System Description. Two rows of carbon steel embed strips are located under each row of tubes as shown in Drawing H-2-119284. The embeds will serve as the welding plates for threaded weld studs to which the tube base assemblies (which are not part of the vault package) will be bolted. The tube base assemblies will not be affixed to the tubes but will restrain the tubes. Each completed alignment fixture will consist of a 2.5-cm (1-in.) carbon steel plate bolted to the embed strips in four places and topped by a short (about 15 cm [6 in.] tall), right-circular cylinder of 1.3-cm (0.5-in.) carbon steel of a diameter slightly larger than the tube and welded to the plate. The tubes will rest on top of the carbon steel plates and be seismically restrained within the 15-cm- (6-in.-) tall cylinder band. (A sketch of the alignment fixture is available.)

4.3.4.3 Functional Requirements. The embeds in vault 1 must remain in place during the DBE to provide horizontal stabilization of the tubes. Functional requirements for embeds in the base slab areas of vaults 2 and 3 were based on seismic loading for the HWVP CSB mission and will require re-evaluation for future uses.

4.3.4.4 System Evaluation. The carbon steel embeds for vault 1 are evaluated for the DBE specified in Section 1.2.2 based on the seismic loading from the storage tubes for the HWVP CSB. Embed-tube interface design will accommodate the current DBE for tubes loaded with MCOs. Any future use of embeds in vaults 2 and 3 must be evaluated against their original design, which was for seismic loading associated with the HWVP CSB mission.

4.3.4.5 Controls (Technical Safety Requirements). Not applicable for this system.

4.4 SAFETY-SIGNIFICANT STRUCTURES, SYSTEMS, AND COMPONENTS

4.4.1 Canister Storage Building Subsurface Structure

4.4.1.1 Safety Function. The CSB subsurface structure provides shielding for the inlet and outlet plenums and for possible future work in adjacent vaults or excavation next to the vaults.

4.4.1.2 System Description. The SNF CSB subsurface structure design includes 1.4-m- (54-in.-) thick reinforced concrete exterior walls and 0.9-m- (36-in.-) thick reinforced concrete interior walls. These wall thicknesses are the same as in the original HWVP CSB design. SNF CSB interior walls were modeled at 1.2 m (48 in.) thickness for structural calculations, but are detailed at 0.9 m (36 in.). This added conservatism to the structural loading of the basemat. The subsurface structure with 0.9-m- (36-in.-) interior walls provides shielding from the source term associated with SNF storage. This source term is substantially less than what would have been present in the HWVP CSB.

4.4.1.3 Functional Requirements. The exterior walls of the subsurface structure were designed to achieve the criteria given in 10 CFR 835, "Occupational Radiation Protection," for uncontrolled access. The radiation field at the exterior surface of the below-grade vault areas when loaded with HWVP CSB sources was calculated to be 0.1 mrem/h. This makes future excavation possible next to vault 3 for installation of additional processes, such as hot vacuum conditioning, if desired.

4.4.1.4 System Evaluation. The shielding calculations show that the radiation field at the exterior walls will be less than 0.05 mrem/h for SNF as the source term. Originally, the interior walls were designed for structural purposes only; shielding calculations now show they provide shielding from the SNF for workers in presently unused vault areas 2 and 3. Interior radiation levels in the presently unused vaults will be about 0.4 mrem/h at the wall (1 cm). Because of the large source configuration, the dose rate inside adjacent vaults (e.g., vault 2) does not decrease very rapidly with distance, remaining at about 0.4 mrem/h for the first meter or so away from the interior wall of vault 2 that is shared with vault 1. Shielding calculation details will be available when the vault package is approved for construction.

4.4.1.5 Controls (Technical Safety Requirements). Not applicable for this system.

4.5 REFERENCES

4.5.1 Documents

Kummerer, M., 1995, *Hazard Category Analysis for the Canister Storage Building*, WHC-SD-SNF-HC-007 Rev. 0, Westinghouse Hanford Company, Richland, Washington.

WHC, 1994, *Hanford Waste Vitrification Plant Canister Storage Building Preliminary Safety Analysis Report Addendum*, WHC-SD-HWV-PSE-001, Rev. 0A, Westinghouse Hanford Company, Richland, Washington.

4.5.2 Drawings

H-2-119278, 1993, *Structural Canister Storage Building Vault Cross Sections*, Rev. 1, Fluor Daniel, Incorporated, Richland, Washington.

H-2-119284, 1993, *Structural Canister Storage Building Vault Canister Base Embeds*, Rev. 1, Fluor Daniel, Incorporated, Richland, Washington.

6.0 PREVENTION OF INADVERTENT CRITICALITY

6.2 REQUIREMENTS

This section lists the design codes, standards, regulations, and DOE orders that are required for establishing the safety basis of the facility. The intent is to provide only the requirements that are specific to Chapter 6.0 and pertinent to the safety analysis.

6.2.1 U.S. Department of Energy Orders and Standards

The following DOE orders and standards are applicable to the safety analysis.

- DOE Order 5480.24, 1992, *Nuclear Criticality Safety*
- DOE Order 420.1, 1995, *Facility Safety* (new).

6.2.2 Industry Consensus Standards

The following industry standards are used by the safety analysts in their calculations.

- ANSI/ANS-8.1-1983, reaffirmed in 1988, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*
- ANSI/ANS-8.7-1975, reaffirmed in 1987, *Guide for Nuclear Criticality Safety in the Storage of Fissile Materials*
- ANSI/ANS-8.10-1983, reaffirmed in 1988, *Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement*
- ANSI/ANS-8.21-1995, 1995, *Use of Fixed Neutron Absorbers in Nuclear Facilities Outside Reactors*.

6.3 ASSESSMENT

MCO containers, loaded with either N Reactor MKIV or MKIA fuel or with scrap pieces, will be stored in the CSB. The design of the CSB is for a maximum capacity of 440 MCO containers, but only 400 MCO containers are planned. No criticality safety issues have been identified in analyses performed to date that indicate the need for design requirements specific to the CSB below-grade structure. Fully-loaded infinite arrays of MCO containers have been analyzed with interspersed moderation. Double-stacking of MCO containers has not been analyzed as the fully-loaded MCO is sufficiently tall (even single-stacked) that additional height does not affect k_{eff} (axial neutron leakage is very small). With full water flooding in the MCO and considering reflection, calculated k_{eff} values are well below 0.95 for MCO containers loaded with intact fuel assemblies. Calculated k_{eff} for the dry

MCO is on the order of 0.33, which is very far below the $k_{\text{eff}} = 0.95$ safety limit. The MCOs that contain some quantity of fuel fragments have been bounded by the calculations performed for rubble following an accident.

Results from criticality analyses performed to date have been reported in Internal Memo 9601, *Preliminary Results: Criticality Analysis of Multiple Canister Overpack Container* (Schwinkendorf 1996b). Additional analysis is ongoing to more fully address accident conditions, including assumptions regarding scrap and potentially more reactive rubble (which could result from the drop accident).

The drop accident initially assumed that all of the fuel breaks apart. Realistic assumptions for rubble (size and packing density) result in a $k_{\text{eff}} < 0.95$ for both the MKIV and MKIA basket designs (Schwinkendorf 1996a).

The bounding analyses described above envelope the specific analyses that will be used to address the CSB vault criticality assessment. The design of the CSB facility itself will not affect these preliminary conclusions.

6.4 REFERENCES

Schwinkendorf, K. N., 1996a, *Criticality Analysis of MCO Container* (Internal Memo 96002 to K. E. Smith, dated January 26), Westinghouse Hanford Company, Richland, Washington.

Schwinkendorf, K. N., 1996b, *Preliminary Results: Criticality Analysis of Multiple Canister Overpack Container* (Internal Memo 9601 to R. G. Cowan, January 10), Westinghouse Hanford Company, Richland, Washington.

14.0 QUALITY ASSURANCE

14.1 INTRODUCTION

This chapter describes the quality assurance program for the SNF CSB Project. The SNF CSB is a nonreactor nuclear facility intended to store MCOs containing SNF. The quality assurance program is founded on 10 CFR 830.120, "Quality Assurance Requirements," and 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," using ASME NQA-1-1989, *Quality Assurance Program Requirements for Nuclear Facilities*, as a guide. This chapter describes the quality assurance program and planned actions that WHC will implement to demonstrate and ensure that the project meets the requirements of 10 CFR 830.120 and 10 CFR 72.

--The WHC SNF CSB quality assurance program plan (QAPP), as referenced in this chapter, shall provide a compliance basis for all CSB Project QAPPs and, with its implementing procedures, shall form the WHC controls for the SNF CSB Project.

14.2 REQUIREMENTS

This section lists the design codes, standards, regulations, and DOE orders that are required for establishing the safety basis of the facility. The intent is to provide only the requirements that are specific to Chapter 14.0 and pertinent to the safety analysis.

14.2.1 U.S. Department of Energy Rules

~~The following rule is applicable to the safety analysis.~~

- ~ 10 CFR 830.120, "Quality Assurance Requirements," *Code of Federal Regulations*.

14.2.2 U.S. Nuclear Regulatory Commission Rules

The following rule is applicable to the safety analysis.

- ~ 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High Level Radioactive Waste," *Code of Federal Regulations*.

14.3 QUALITY ASSURANCE PROGRAM ORGANIZATION

14.3.1 Organization

This section describes the organizational responsibilities and authorities for the SNF CSB Project. The assignment of responsibilities is

based on the policy that line organizations achieve quality and the quality organization provides overview to verify achievement of quality.

Several organizations contribute to the SNF CSB. The design and construction of the project represent an integrated effort by DOE, the owner; WHC, the design authority and maintenance and operations contractor; Fluor Daniel, Inc., the architect-engineer, serving as the design agent; ICF Kaiser Hanford (ICF KH), the construction manager; and Mowat Construction Company, the constructor. The involvement of numerous participants underscores the need to effectively integrate activities to reduce or eliminate redundancies. The problem of overlap in responsibilities has been addressed by adopting an integrated management team approach. An organization chart depicting the interfaces of the participants is shown in Figure 14.3-1.

Figure 14.3-1. Spent Nuclear Fuels Canister Storage Building Project Organization.

Specific detail regarding the general roles and responsibilities of the participants, delegation of work, stop-work authority and management process is provided in WHC-SD-W379-QAPP-001, *Quality Assurance Program Plan for Project W-379, Spent Nuclear Fuels Canister Storage Building Project* (WHC 1995a), which was approved by DOE's Richland Operations Office DOE-RL).

14.3.2 Program

WHC has an established quality assurance program that is based on 10 CFR 830.120. The WHC quality assurance program for this project is derived from the applicable sections of 10 CFR 830.120 and 10 CFR 72, using ASME NQA-1-1989, including the supplements, as a guide. In response to the *Price-Anderson Amendments Act of 1988* and the Quality Assurance Rule, (10 CFR 830.120), WHC has issued WHC-SP-1131, *The Westinghouse Hanford Company Quality Assurance Program and Implementation Plan*. That document serves as a basis for WHC implementation of 10 CFR 830.120 and is the foundation for extending implementation requirements to contractors and subcontractors.

Specific details regarding reporting, independence of personnel, planning, graded quality assurance, personnel selection, qualification, training and indoctrination, management assessments, and procedures are presented in Chapter 2 of WHC-SD-W379-QAPP-001 (WHC 1995a).

14.4 QUALITY IMPROVEMENT

This section describes WHC programs and processes used to correct adverse conditions affecting quality. Specifically included in this section is a discussion of the control and disposal of nonconforming material, parts, and components.

14.4.1 Control of Nonconforming Items

Nonconformance report (NCR) procedures are used to promote consistency, timeliness of problem resolution, and processing of the necessary documentation. Items or practices that do not conform to specified requirements, or whose conformance is indeterminate, shall be documented on an NCR and controlled to prevent inadvertent installation or use. Controls for nonconformance are provided for notification of affected organizations; for identification of affected organizations; and for identification, documentation, evaluation, segregation, (when practical), and disposition. Nonconformance shall be tracked to closure to ensure that approved dispositions are properly documented. All personnel associated with the project are responsible for documenting instances of nonconformance and for seeking resolution.

Specific detail of the activities and controls for reporting nonconforming material, parts, components, or conditions of the project are presented in Chapter 15 of WHC-SD-W379-QAPP-001 (WHC 1995a).

14.4.2 Corrective Action

The organization that detects a condition adverse to safety is responsible for initiating action to promptly identify and document the condition. If the adverse condition involves an item or construction practice that is described by released design documents, an NCR is initiated and processed in accordance with approved procedures. If the adverse condition is discovered as a result of an audit or surveillance, an audit finding/observation or inspection/surveillance report is initiated and processed. If the condition is significantly adverse to quality, a corrective action request or stop-work order is initiated and processed.

Specific detail regarding trend analysis, corrective action criteria, tracking and remedial action for conditions adverse to safety is presented in Chapter 16 of WHC-SD-W379-QAPP-001 (WHC 1995a).

14.4.3 Control of Nonconformance for Purchased Items

The SNF CSB quality assurance program requires that suppliers develop procedures for the disposition of items that do not meet procurement requirements. Specific requirements regarding the development of these procedures is contained in Chapter 7 of WHC-SD-W379-QAPP-001 (WHC 1995a).

14.5 DOCUMENTS AND RECORDS

This section describes the document control and records management program associated with quality assurance.

14.5.1 Document Control

An integrated document control system for the design and construction phases of the project is in place. This system provides a single focus (i.e., SNF CSB management) for document control practices for all participants. This system requires all participants to develop and implement procedures that ensure the program documents affecting quality shall be prepared, revised, reviewed, approved, and issued in a prescribed and controlled manner.

Specific detail regarding the process for document control for the SNF CSB Project is presented in Chapter 6 of WHC-SD-W379-QAPP-001 (WHC 1995a).

14.5.2 Records Management

The activities and controls for the control of records for the SNF CSB Project are presented in Chapter 17 of WHC-SD-W379-QAPP-001 (WHC 1995a). Specific detail is presented regarding the establishment of a record system, record administration, record validation, retention, receipt, corrective information in records, and storage, preservation, and safekeeping of records.

14.6 QUALITY ASSURANCE PERFORMANCE

This section provides an overview of the process that the SNF CSB Project will use to ensure that the performed work meets the project requirements.

14.6.1 Work Processes

This section discusses WHC management programs that ensure performance of tasks under controlled conditions, with applicable calibrated instrumentation and in accordance with established technical standards and administrative controls.

14.6.1.1 Control of Process. This section describes the measure established to control special processes and to ensure that they are accomplished by qualified personnel using written procedures qualified in accordance with applicable codes, standards, specifications, or other special requirements. Also described are those measures that ensure qualifications of special processes, personnel performing special processes, and equipment are kept current, and that the associated records are maintained.

Specific detail regarding these measures are provided in Chapter 9 of WHC-SD-W379-QAPP-001 (WHC 1995a).

14.6.1.2 Control of Measuring and Test Equipment. Each organization responsible for using measuring and test equipment shall use controlled procedures for calibration and control of that equipment. Any measuring and test equipment used for acceptance testing, verification, or data collection is subject to these controls. These controls provide for calibration and adjustment at specified intervals to maintain accuracy within necessary limits.

The specific details that describe the activities and controls for measuring and test equipment for the SNF CSB Project are contained in Chapter 12 of WHC-SD-W379-QAPP-001 (WHC 1995a).

14.6.2 Design

This section describes how quality assurance is integrated into the design activities.

14.6.2.1 Design Control. The design of the SNF CSB shall be defined, controlled, and verified. Applicable design inputs are specified on a timely basis and correctly translated into design documents. Design interfaces are identified and controlled. Design adequacy is verified by persons other than those who designed the item. Design changes including field changes, are governed by control measures commensurate with those applied to the original design.

Specific detail regarding the design control process, change control, interface control, documentation and records, technical reviews, peer reviews, and computer software control are presented in Chapters 3 and 19 of WHC-SD-W379-QAPP-001 (WHC 1995a).

14.6.3 Procurement

This section describes how quality assurance is integrated into the procurement process. The section also describes how prospective suppliers are evaluated, selected and their acceptability monitored.

14.6.3.1 Procurement Document Control. The project quality assurance program controls the process of developing and using procurement documents to ensure that requirements for quality are formally communicated to vendors of items and services. The Construction Manager is given responsibility for management and administration of construction subcontracts. The applicable technical requirements will be included in the procurement documents. The construction manager reviews the procurement document to ensure that all appropriate requirements are addressed. The procurement document is then controlled so that subsequent changes to the document are subjected to the same level of review as the original. After review, the procurement document is approved and released to the vendors. Any changes to the procurement documents after release shall be approved in the same manner as the original design approval and procurement approvals.

Suppliers and subcontractors shall have quality assurance programs consistent with the requirements contained in the procurement documents. Suppliers of materials shall meet the quality requirements specified in the specification.

Specific details regarding the procurement document control including document contents, document reviews, change control, and quality assurance overview of procurement activities is contained in Chapter 4.0 of WHC-SD-W379-QAPP-001 (WHC 1995a).

14.6.3.2 Control of Purchased Items and Services. This section describes the controls applied to the purchase of items and services that affect quality. Implementation of these controls ensures conformance with specified requirements. As appropriate, these controls provide for source evaluation and selection, evaluation of objective evidence of quality furnished by the supplier, source inspection, audit, and examination of items or services upon delivery or completion.

Specific details regarding the description of the controls applied to the purchase of items and services is provided in Chapter 7 of WHC-SD-W379-QAPP-001 (WHC 1995a).

14.6.4 Inspection and Testing for Acceptance. This section describes how quality assurance is integrated into inspection and testing of programs.

14.6.4.1 Inspection. This section describes the controls for inspections required of the SNF CSB project to verify conformance of items or activities to specified requirements.

Construction inspections, control practices, characteristics to be inspected, and methods to be employed are described in the work procedures and inspection planning documents. Inspection activities and results are

documented. The WHC projects quality assurance group is responsible for monitoring construction inspection activities by surveillance.

Title III inspection requirements will be performed for construction workmanship, materials, and equipment for conformance to approved drawings and specifications.

Specific detail regarding personnel, hold points, inspection planning, in-process inspection and final inspections are provided in Chapter 10 of WHC-SD-W379-QAPP-001 (WHC 1995a).

14.6.4.2 Test Control. This section provides a description of the activities and controls for testing control at the project. Test requirements and acceptance criteria for inclusion in the procurement and construction specifications are prepared by Fluor Daniel, Inc., the architect-engineer. These tests are executed to provide verification of conformance to requirements. Final detailed test plans and/or acceptance test procedures are prepared by ICF KH, the construction manager, and vendors/subcontractors in accordance with the requirements of the procurement and construction specifications and vendor data.

Test plans and procedures identify characteristics to be tested and methods to be used. These detailed test plans and/or acceptance test procedures will be reviewed and approved by the architect-engineer, construction manager, and/or WHC as defined in the design media.

All tests shall be conducted using approved procedures and/or detailed test plans, checklists, and trained personnel. Data collected as a result of these tests are reviewed and evaluated for acceptability.

Specific detail regarding test requirements, test procedures, test results and test records are presented in Chapter 11 of WHC-SD-W379-QAPP-001 (WHC 1995a).

14.6.5 Independent Assessment

WHC shall establish requirements for independent assessments to provide independent verification of status, compliance, and implementation effectiveness of this quality assurance program and its elements.

Specific details of the provisions for implementing independent assessments are provided in Chapter 18 of WHC-SD-W379-QAPP-001 (WHC 1995a).

14.6.5.1 Westinghouse Hanford Company Quality Assurance Assessment for Construction Restart. WHC has initiated a multiphase program to assess the adequacy of the construction activities associated with the as-completed portion of the HWVP. The purpose of this program is to provide reasonable assurance to WHC that adequate quality assurance exists regarding the as-completed work to warrant restart of construction activities to complete the CSB subsurface construction. The phases of the program include (1) comparison of HWVP Project quality assurance program to 10 CFR 830.120, (2) preparation of QAPPs for CSB activities that are in accordance with the

requirements of 10 CFR 830.120, (3) Fluor Daniel, Inc., physical walkdown inspection of existing CSB construction, (4) comprehensive review of existing construction records, and (5) CSB construction restart readiness assessment.

14.6.5.1.1 Westinghouse Hanford Company Quality Assurance Assessment for Hanford Waste Vitrification Project.

Purpose and Scope. *Westinghouse Hanford Company Quality Assurance Assessment, Hanford Waste Vitrification Project, Comparison of HWVP Project Quality Assurance Program to Title 10 of Federal Regulations 830.120* (WHC 1995b), records an assessment performed of the programmatic controls associated with the development of the HWVP CSB project to date. The scope of the assessment was to identify the level of effective implementation of the quality assurance program controls according to the intent of 10 CFR 830.120

-- It is necessary to understand that the HWVP project was "ramped down" and deferred to standby/mothball status in October 1993. All project activities, except technology development and records closure, were discontinued at that time. This assessment addresses the programmatic controls applied until then.

Summary of Compliance Assessment Results. The results of this assessment were very positive. All contractors assessed (WHC, Fluor Daniel, inc., and United Engineers and Constructors-Catalytic) had quality assurance programs in place that generally met the intent of 10 CFR 830.120. Deficiencies noted during this assessment were also detected by the contractor's quality assurance programs. This is a positive indication that the quality assurance programs were effective. General conditions that were noted and have been resolved are

- o The status of in-work project records.....
- o ~~The difficulty identifying and retrieving records.....~~
- o The presence of unauthorized and/or uncontrolled design documents in the field.

The assessment summary and conclusions are contained in the Quality Assurance Assessment (WHC 1995b, pp 6-9).

14.6.5.1.2 Preparation of Quality Assurance Program Plans for the Canister Storage Building. QAPPs have been prepared in accordance with 10 CFR 830.120 and 10 CFR 72. The following list indicates each member of the SNF CSB Project team and the corresponding reference for its respective QAPP.

- o WHC-ICF KH

Quality Assurance Program Plan for Project W-379, Spent Nuclear Fuels Canister Storage Building Project, WHC-SD-W379-QAPP-001 (WHC 1995a), the project QAPP document

- Fluor Daniel, Inc.

Detailed Design Canister Storage Building (CSB) Quality Assurance Program Plan (Fluor 1995)

- Mowat Construction Company

Quality Assurance Manual (Mowat 1996)

Both the Fluor Daniel, Inc., and the Mowat Construction Company QAPPs have been reviewed and approved by the WHC Projects Quality Assurance group in accordance with the WHC-ICF KH QAPP (WHC 1995a). This WHC procedure applies to supplier evaluation for all subcontractor quality plans.

14.6.5.1.3 Fluor Daniel, Incorporated, Physical Construction Inspection. Fluor Daniel, Inc., is performing a physical inspection of the existing CSB construction (the original HWVP) completed to date. The purpose of this inspection is to review as-built construction for any deterioration or degradation of the in-place construction against the design drawings to ensure that the facility is ready for construction restart activities.

This assessment will be completed before construction restart, and any noted deficiencies will be documented and dispositioned (i.e., plan in place) by the time of construction restart.

14.6.5.1.4 Quality Assurance Record Review. The WHC Projects Quality Assurance group is conducting a review of all construction records related to the CSB Project. The purpose of this review is to ensure that all known deficiencies have been identified and subsequently dispositioned and to determine that records exist of the existing structure to support restart. The types of documents being reviewed are supplier/subcontractor disposition requests, UCAT deficiency reports, corrective action reports, open items lists, contractor surveillance/inspection reports, and UCAT surveillance/inspection reports. The organizations whose records are being reviewed are DAMCO (construction), Morrison (excavation), Central Premix (concrete), CHEN Northern (testing), and Century West (testing). Any deficiencies noted during the assessment will be dispositioned (i.e., plan in place) before the restart of construction.

A previous examination of the construction records (completed before preparing the construction bid specification) identified nine deviations that required disposition (three regarding repair of concrete surfaces, two regarding rebar repair, two regarding shrinkage crack repair, and two regarding concrete rock pocket repair). All of the items were incorporated in the construction bid specification and will be corrected as part of the restart activity.

14.6.5.1.5 Canister Storage Building Construction Readiness Assessment. Preparation for restart of construction on the CSB involves many organizations and considerations. Report No. 951108-001, *Canister Storage Building Construction Restart Readiness Assessment* (Ares 1996), identifies activities to be completed before restarting construction of the CSB. The assessment provides a mechanism for managing the completion and review of documentation to complete the required activities.

The key to successful initiation of major construction activities, and particularly to restart of construction of the first package (vault construction to elevation 704 ft), will be the ability of WHC, ICF KH, and DOE-RL personnel to successfully manage completion of the prerequisite activities and to concisely and effectively communicate to WHC and DOE-RL management the reasons why restart is appropriate. To accomplish these efforts, a three-step process will be used.

1. Prerequisite determination

A list is developed of activities that must be completed before construction restart. This master list identifies the deliverable associated with each activity and the organization responsible for completion. This list contains requirements from the areas of environmental permitting, design, construction, and project management. The requirements are developed from laws, regulations, orders, procedures, and best industry practices. The determination of prerequisites is described in Section 3.2 of Report No. 951108-001 (Ares 1996).

2. Prerequisite completion/review

Completion of each deliverable item identified as a prerequisite is accomplished by the appropriate organization. Detailed schedules are developed during this step (if required), regular status meetings are held, and issues are resolved. Alternatives to the original prerequisites may be proposed, evaluated, and implemented. The deliverables developed during this process must be subjected to a two-part review. First, a technical review for accuracy and completeness is accomplished according to existing policies and procedures. The second part is an "adequacy for construction restart" review. The purpose of this review is to determine whether the product fills the needs to support restart of construction. This review is accomplished as described in Section 3.3 of Report No. 951108-001 (Ares 1996).

3. Assessment/documentation of readiness for restart.

Preparations for restart of construction are independently evaluated. The breadth of the assessment is the prerequisite list. Upon completion of the assessment, a formal report is written. This report will document the results of the assessment, identify any areas of concern or open items, and make a formal recommendation to WHC and DOE-RL management on whether or not the preparations for restart of construction are adequate. This effort is accomplished as described in Section 3.4 of Report No. 951108-001 (Ares 1996).

14.6.6 Conclusion

Based on the acceptable results of a thorough, multiphase, comprehensive quality assessment of the existing HWVP structures and the QAPP currently in place for the CSB, an appropriate level of quality assurance is being demonstrated to warrant restart of the subsurface construction of the CSB.

14.7 REFERENCES

- Ares, 1996, *Canister Storage Building Construction Restart Readiness Assessment*, Report No. 951108-001, Ares Corporation, Richland, Washington.
- Fluor, 1995, *Detailed Design Canister Storage Building (CSB) Quality Assurance Program Plan*, Fluor Daniel, Incorporated, Richland, Washington.
- Mowat, 1996, *Quality Assurance Manual*, Rev. 1, Mowat Construction Company, Kirkland, Washington.
- WHC, 1995a, *Quality Assurance Program Plan for Project W-379, Spent Nuclear Fuels Canister Storage Building Project*, WHC-SD-W379-QAPP-001, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1995b, *Westinghouse Hanford Company Quality Assurance Assessment, Hanford Waste Vitrification Project, Comparison of HWVP Project Quality Assurance Program to Title 10 of Federal Regulations 830.120*, Westinghouse Hanford Company, Richland, Washington.

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16.0 PROVISIONS FOR DECOMMISSIONING AND DECONTAMINATION

16.2 REQUIREMENTS

This section lists the design codes, standards, regulations, and DOE orders that are required for establishing the safety basis for the CSB. The intent is to provide only the requirements that are pertinent to the safety analysis.

16.2.1 U.S. Department of Energy Orders and Standards

The following DOE order is applicable to the safety analysis.

- - ^ DOE Order 6430.1A, 1989, *General Design Criteria*. This Order establishes general requirements for decontamination and decommissioning in Division 13, "Special Facilities," Section 1300, "General Requirements," paragraph 1300-11, "Decontamination and Decommissioning," and Section 1320, "Irradiated Fissile Material Storage Facilities," paragraph 1320-7, "Decontamination and Decommissioning." In particular, paragraph 1300-11 requires that "design of the areas in a facility that may become contaminated with radioactive. . . materials. . . shall incorporate measures to simplify future decontamination." Similarly, paragraph 1320-7 requires that "the facility design shall include features that will facilitate decontamination for future decommissioning." The spread of contamination to the vault is not postulated as a result of normal operation or accident conditions. The MCOs are vented through a HEPA filter to the storage tube during the staging period. The storage tube could become contaminated from failure of the HEPA filter or residual contamination on the exterior of the MCO. The storage tube is sealed from the vault but could release contamination to the operating deck during accident conditions. The operating deck is sealed from the vault and is ventilated through a HEPA filter. Therefore, contamination of the vault caused by routine operations or accident conditions is not considered and the below-grade portion of the vault needs no special provisions to facilitate decontamination activities.

16.2.2 U.S. Nuclear Regulatory Commission Rules and Guidance

The following NRC rules and guidance are applicable to the safety analysis.

- ^ 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Subpart F, "General Design Criteria," paragraph 72.130, "Criteria for Decommissioning," *Code of Federal Regulations*. This rule requires that ISFSIs be designed to facilitate decontamination at the time of decommissioning. As noted in Section 16.2.1, provisions for decontamination of the vault are not required.

- ~ Regulatory Guide 3.48, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)*. Section 9.6 of the guide requires that the safety analysis report describe plans for facilitating the decommissioning process and how those plans have been used in designing the facility. The safety analysis report for the CSB will contain a section describing any design features that facilitate decontamination and decommissioning. As noted in Section 16.2.1, provisions for decontamination of the vault are not required.

16.2.3 Industry Consensus Standards and Other Documentation

The following industry standard is applicable to the safety analysis.

- ~ ANSI/ANS-57.9-1992, 1992, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)*, Subsection 6.14, "Decommissioning." This industry standard requires that the design of ISFSIs incorporate features that facilitate decontamination and decommissioning. As noted in Section 16.2.1, provisions for decontamination of the vault are not required.

17.0 MANAGEMENT, ORGANIZATION, AND INSTITUTIONAL SAFETY PROVISIONS

17.1 INTRODUCTION

This chapter presents information on management, technical, and other organizations that support safe operation. Included in this chapter will be a description of the overall structure of the organizations and personnel with responsibilities for facility safety and the interfaces between those organizations. Also described are the programs that promote safety consciousness and morale, including safety culture, performance assessment, configuration and document control, occurrence reporting, and staffing and qualification.

17.2 REQUIREMENTS

This section lists the design codes, standards, regulations, and DOE orders that are required for establishing the safety basis of the facility. The intent is to provide only the requirements that are specific for Chapter 17.0 and pertinent to the safety analysis.

17.2.1 U.S. Department of Energy Orders and Rules

The following DOE orders, standards, and rules are applicable to the safety analysis.

- DOE Order 4700.1, 1987, *Project Management System*
- DOE Order 5400.5, 1993, *Radiation Protection of the Public and the Environment*
- DOE Order 5480.9A, 1994, *Construction Project Safety and Health Management*
- DOE Order 5480.10, 1985, *Contractor Industrial Hygiene Program*
- DOE Order 5480.21, 1991, *Unreviewed Safety Questions*
- DOE Order 5480.22, 1992, *Technical Safety Requirements*
- DOE Order 5480.23, 1992, *Nuclear Safety Analysis Reports*
- DOE-STD-1073-93, 1993, *Guide for Operational Configuration Management Program: Including the Adjunct Programs of Design Reconstitution and Material Condition and Aging Management*
- 10 CFR 830.120, "Quality Assurance Requirements," *Code of Federal Regulations*.

17.2.2 U.S. Nuclear Regulatory Rules and Guidance

The following NRC rules and guidance are applicable to the safety analysis.

- ~ 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High Level Radioactive Waste," *Code of Federal Regulations*.

17.2.3 Consensus Industry Standards

The following industry standard is applicable to the safety analysis.

- ~ ASME NQA-1-1989, 1989, *Quality Assurance Program Requirements for Nuclear Facilities*, with current addenda.

17.3 ORGANIZATIONAL STRUCTURE, RESPONSIBILITIES, AND INTERFACES

This section summarizes the overall structure of the SNF Project organization. Also included is a discussion of the separate and distinct entities that are organized into a safety-conscious and responsive organization to ensure and enhance facility safety.

17.3.1 Organizational Structure

The CSB Project team consists of the DOE Spent Nuclear Fuels Project Division, WHC-ICF KH, Fluor Daniel, Inc., and other subcontractors. The organizational relationships of the SNF CSB Project are shown in Figure 17.3-1. The overall responsibility matrix is included in Appendix A of WHC-SD-W379-PMP-001, *Project W-379, Canister Storage Building Project Management Plan, January 26, 1996* (WHC 1996a). References to project participants include the full project team: the DOE Spent Nuclear Fuels Project Division, WHC-ICF KH, Fluor Daniel, Inc., and other subcontractors. References to contractor participants include the project participants except the DOE Spent Nuclear Fuels Project Division.

17.3.2 Organizational Responsibilities

The CSB Project's functions and performing organizations are shown on Figure 17.3-2. Specific details regarding the organization's responsibilities and the interfaces between the various contractors are presented in Section 3.2, "Responsibilities" (pages 5-14), of WHC-SD-W379-PMP-001 (WHC 1996a).

Figure 17.3-1. Spent Nuclear Fuel Canister Storage Building Project's Organizational Relationships.

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Figure 17.3-2. Canister Storage Building Project
Functions and Performing Organizations.

17.3.3 Staffing and Qualifications

This section summarizes the bases for the staffing levels and the knowledge, skills, and abilities of facility personnel in the organizations covered in this chapter. Also discussed in this chapter are the programs and provisions for monitoring the safety performance of the staff.

17.3.3.1 Project Staffing. Project staffing is included as part of the resource-loaded, overall project integrated schedule that was prepared as part of the multiyear program plan for the SNF Project.

17.3.3.2 Qualification and Training. The CSB Project will implement a systematic approach to performance-based training that identifies the training necessary to meet requirements for assigned staff. The performance-based training is structured to meet the guidelines found in WHC-CM-6-12, *Project Department Procedures*, P-9, "Projects Department Training and Indoctrination." CSB Project contractor participants, including subcontractors, will be required to submit their qualification and training plans to WHC, upon request, for review and approval. WHC is responsible for the qualification, as well as the planning and scheduling, of all CSB Project contractor participants.

17.3.3.3 Safety Performance. Section 6.2, "Safety Authorization Process," pages 28-31, of WHC-SD-W379-PMP-001 (WHC 1996a) identifies the major safety documents that will be required to implement the recommended path forward for the CSB. It includes the requirements basis, the scope of the safety document, the review and authorization approach for new facilities, and the application of NRC technical requirements to new SNF Project facilities to ensure nuclear safety equivalence with comparable NRC-licensed facilities. The documents that collectively make up the safety basis will be identified. These documents will be maintained over the lifetime of the facility.

17.4 SAFETY MANAGEMENT POLICIES AND PROGRAMS

The following section identifies and describes programs to enhance facility safety.

17.4.1 Safety Review and Performance Assessment

This section summarizes the programs and procedures used to ensure independent oversight, safety review, determination of unreviewed safety questions, and appraisal of the safety performance of all of the organizations involved in the management of safety, such as industrial safety, fire inspections, and hazardous material control.

17.4.1.1 Independent Oversight and Safety Review. As discussed in WHC-SD-SNF-PLN-012, *Spent Nuclear Fuel Project Safety Management Plan* (WHC 1996b), the safety analysis report will be reviewed by the appropriate WHC functional disciplines, including the WHC Independent Safety Group. Review comments received at the WHC functional level and higher will be formally dispositioned using Review Comment Record forms. Following the WHC functional review, the phased safety analysis report documentation will be

forwarded to the Safety and Environmental Advisory Council for review. Again, review comments received will be formally dispositioned using Review Comment Record forms. Figures 17.4-1 and 17.4-2 depict the flow process that will be followed for the preparation, review, and approval of the safety analysis report.

17.4.1.2 Identifying and Resolving Unreviewed Safety Questions. The unreviewed safety question process exists to allow contractors to make physical and procedural changes and conduct tests and experiments without prior DOE approval as long as the changes do not explicitly or implicitly affect the safety basis of the facility or result in a change to a technical safety requirement (or to technical specifications or operational safety requirements). The unreviewed safety question process also ensures that discoveries of conditions outside the authorization basis or technical safety requirements are identified and actions taken to modify the conditions or to redefine the safety basis.

The implementing procedure for the unreviewed safety question process described in DOE 5480.21 is WHC-CM-1-5, *Standard Operating Practices*, Section 7.3, "Identifying and Resolving Unreviewed Safety Questions," which also provides guidance for preparing justification to continue operations and for determining when they are required. This procedure requires the establishment of a facility plant review committee, which is a good management practice.

17.4.1.3 Industrial Safety and Hazardous Material Control. The WHC program requirements for establishing and carrying out the Occupational Safety and Health Administration's *Hazardous Waste Operations and Emergency Response Standard* as required by the DOE is presented in WHC-CM-4-40, *Industrial Hygiene Manual*, Section 2.19, "Health and Safety for Hazardous Waste Field Operations."

WHC policy ensures that its employees are provided with adequate protection, training, and information about hazardous waste and emergency response operations to conduct these operations safely. This policy also is applicable to hazardous waste operations performed by other contractors at WHC-managed sites.

17.4.2 Configuration and Document Control

17.4.2.1 Configuration Control. WHC is responsible for the overall development, implementation, and maintenance of the baseline documentation. These responsibilities will be executed in accordance with WHC-CM-6-2. WHC will ensure that configuration management activities and systems engineering activities are performed while better defining and controlling the project baseline and associated documentation. These activities will be applied to all systems and subsystems necessary to achieve all functional requirements and deliver all products to satisfy the integrated technical baseline and overall project objectives. At all times during the life of the project, current configuration of the project records and files will include conceptual and definitive systems designs, system and material inspection reports, test reports, operating and surveillance procedures, and vendor documentation.

Figure 17.4-1. Phased Safety Analysis Report Preparation and Review Process.

Figure 17.4-2. Final Safety Analysis Report Review and Approval Process.

Configuration management maintains and controls changes to the technical baseline once the baseline is placed under change control. Configuration management performed will be consistent with Chapter III of DOE Order 4700.1, the Hanford Site Configuration Management Plan, and WHC-SD-SNF-CM-001, *Spent Nuclear Fuel Project Configuration Management Plan* (WHC 1995a), as well as applicable portions of DOE-STD-1073-93.

Figures 17.4-1 and 17.4-2 define the approval authority for changes to the defined technical baseline.

17.4.2.2 Document Control. The currently available systems and processes for implementing and maintaining an effective document control and records management program are described in WHC-SD-SNF-MP-001, *Spent Nuclear Fuel Project Document Management Plan* (WHC 1995b). This program governs the methods by which documents are generated, released, distributed, maintained current, retired, and ultimately disposed of. The plan outlines the documentation needs of the SNF Project, identifies existing processes currently provided by Documentation and Records Management that can satisfy those needs, and documents the strategy for ensuring the SNF Project has in place a fully functioning and integrated document control and records management program.

Because of the recent consolidation of the previously independent document control and records management functions from BCS Richland, Inc., ICF KH, and WHC, BCS Richland, Inc., is now able to provide better integrated and significantly enhanced document services to Hanford Site contractors. More specifically, the document management capabilities now available to the SNF Project represent the composite strengths of the individual programs. The best of each contractor's processes was merged into a common process. Management of the new process by a single organization will benefit the SNF Project and other users.

17.4.3 Occurrence Reporting

The section describing WHC programs and processes used to correct adverse conditions affecting quality is presented in Section 14.4.

17.4.4 Safety Culture

The Safety Management Plan (WHC 1996b) describes the nuclear safety regulatory requirements basis for the SNF Project and establishes the plan to achieve compliance with the regulatory requirements. Compliance will be accomplished by establishing the proper authorization basis for each of the SNF Project's processes and facilities through development and documentation of the appropriate design bases and safety analyses that comply with the regulatory requirements.

The Safety Management Plan (WHC 1996b) defines responsibilities for the preparation, review, and approval of the safety analysis documentation, and it establishes the approach for integrating the safety analysis, safety documentation, and independent safety reviews with the SNF Project's design, construction, and startup activities. The Safety Management Plan (WHC 1996b)

also defines the SNF Project's organizational relationships to regulatory compliance and safety management during the design, construction, and preoperational phases of the project.

17.5 REFERENCES

WHC, 1995a, *Spent Nuclear Fuel Project Configuration Management Plan*, WHC-SD-SNF-CM-001, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

WHC, 1995b, *Spent Nuclear Fuel Project Document Management Plan*, WHC-SD-SNF-MP-001, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

WHC, 1996a, *Project W-379, Canister Storage Building Project Management Plan*, January 26, 1996, WHC-SD-W379-PMP-001, Rev. 0, Westinghouse Hanford Company, Richland, Washington

WHC, 1996b, *Spent Nuclear Fuel Project Safety Management Plan*, WHC-SD-SNF-PLN-012, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-1-5, *Standard Operating Practices*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-4-40, *Industrial Hygiene Manual*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-6-2, *Project Management*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-6-12, *Project Department Procedures*, Westinghouse Hanford Company, Richland, Washington.