sta 4. (2)

MAY 21 1997 ENGINEERING DATA TRANSMITTAL

Page 1 of 1

1. EDT 607687

2. To: (Receiving Organization) Distribution		3. From: (Originating Organization) Nuclear Safety		4. Related EDT No.: 607687	
5. Proj./Prog./Dept./Div.: Spent Nuclear Fuel Project		6. Design Authority/ Design Agent/Cog. Engr.: L. J. Garvin III		7. Purchase Order No.: N/A	
8. Originator Remarks: This document is being distributed for approval and release.				9. Equip./Component No.: N/A	
11. Receiver Remarks: 11A. Design Baseline Document? ☒ Yes ☒ No 6/6/97 5/21/97				10. System/Bldg./Facility: CSB/CVD	
				12. Major Assm. Dwg. No.: N/A	
				13. Permit/Permit Application No.: N/A	
14. Required Response Date: 05/09/97					

15. DATA TRANSMITTED					(F)	(G)	(H)	(I)
(A) Item No.	(B) Document/Drawing No.	(C) Sheet No.	(D) Rev. No.	(E) Title or Description of Data Transmitted	Approval Designator	Reason for Transmittal	Originator Disposition	Receiver Disposition
1	HNF-SD-SNF-SP-012		0	Additional Guidance for Including Nuclear Safety Equivalency in the Canister Storage Building and Cold Vacuum Drying Facility Final Safety Analysis Report	S,Q	1-2	1-2	

16. KEY					
Approval Designator (F)		Reason for Transmittal (G)			Disposition (H) & (I)
E, S, Q, D or N/A (see WHC-CM-3-5, Sec.12.7)		1. Approval 2. Release 3. Information	4. Review 5. Post-Review 6. Dist. (Receipt Acknow. Required)		1. Approved 2. Approved w/comment 3. Disapproved w/comment 4. Reviewed no/comment 5. Reviewed w/comment 6. Receipt acknowledged

17. SIGNATURE/DISTRIBUTION
(See Approval Designator for required signatures)

(G) Reason	(H) Disp.	(J) Name	(K) Signature	(L) Date	(M) MSIN	(G) Reason	(H) Disp.	(J) Name	(K) Signature	(L) Date	(M) MSIN
1-2		Design Authority	N/A			1	1	for A. Smith		5/15/97	R3-15
1-2		Design Agent	N/A								
1-2	1	Cog. Eng.	L. J. Garvin	5/15/97	63-15						
1-2	1	Cog. Mgr.	R. P. Ombreg	5/15/97	63-15						
1-2	1	QA	B. R. Fillion	5/15/97	67-41						
1-2	1	Safety	T. D. Merkle	5/15/97	63-79						
1-2	1	QA	T. D. Hays	5/15/97	58-06						

18. Signature of EDT Originator L. J. Garvin 5/15/97		19. Authorized Representative Date for Receiving Organization		20. Design Authority/ Cognizant Manager R. P. Ombreg 5/19/97		21. DOE APPROVAL (if required) Ctrl. No. ☐ Approved ☐ Approved w/comments ☐ Disapproved w/comments	
---	--	---	--	---	--	--	--

Additional Guidance for Including Nuclear Safety Equivalency in the Canister Storage Building and Cold Vacuum Drying Facility Final Safety Analysis Report

L. J. Garvin

Fluor Daniel Hanford, Inc., Richland, WA 99352

U.S. Department of Energy Contract DE-AC06-96RL13200

EDT/ECN: 607687

UC: 620

Org Code: 2T640

Charge Code: LG070

B&R Code: EW31354

Total Pages: 39

376 W 70400 Rms 4/6/97

Key Words: Nuclear, Safety, Equivalency, Canister Storage Building,
Cold Vacuum Drying Facility, Analysis

Abstract: This document provides guidance for the production of safety
analysis reports that must meet both DOE Order 5480.23 and STD 3009, and
be in compliance with the DOE regulatory policy that imposes certain NRC
requirements.

TRADEMARK DISCLAIMER. Reference herein to any specific commercial product, process, or service by
trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its
endorsement, recommendation, or favoring by the United States Government or any agency thereof or
its contractors or subcontractors.

Printed in the United States of America. To obtain copies of this document, contact: Document
Control Services, P.O. Box 950, Mailstop H6-08, Richland WA 99352, Phone (509) 372-2420;
Fax (509) 376-4989.

[Signature]
Release Approval

5/20/97
Date

Approved for Public Release

**ADDITIONAL GUIDANCE FOR INCLUDING NUCLEAR SAFETY EQUIVALENCY IN THE
CANISTER STORAGE BUILDING AND COLD VACUUM DRYING FACILITY
FINAL SAFETY ANALYSIS REPORTS**

**HNF-SD-SNF-SP-012
Revision 0**

May 1997

This page intentionally left blank.

CONTENTS

1.0	INTRODUCTION	1
2.0	BACKGROUND	1
3.0	IMPLEMENTATION	3
3.1	CANISTER STORAGE BUILDING	3
3.2	COLD VACUUM DRYING FACILITY	3
4.0	REFERENCES	5

LIST OF TABLES

1	Items in Regulatory Guide 3.48 that are Unique to the Storage of Spent Nuclear Fuel	T-1
2	Items in Title 10, <i>Code of Federal Regulations</i> , Section 72.24, that are Unique to the Storage of Spent Nuclear Fuel	T-9
3	Information to be Added to the Final Safety Analysis Report for the Canister Storage Building that is not Required by DOE Order 5480.23 or by DOE-STD-3009-94 but is Needed to Implement Nuclear Safety Equivalency	T-11
4	Items of Regulatory Guide 3.26 that are Unique to Fuel Conditioning	T-19
5	Information to be Added to the Final Safety Analysis Report for the Cold Vacuum Drying Facility that is not Required by DOE-STD-3009-94 and that is Unique to the Conditioning Process	T-25

LIST OF TERMS

CSB	Canister Storage Building
CVDF	Cold Vacuum Drying Facility
DOE	U.S. Department of Energy
FSAR	final safety analysis report
NRC	U.S. Nuclear Regulatory Commission
SNFP	Spent Nuclear Fuel Project

ADDITIONAL GUIDANCE FOR INCLUDING NUCLEAR SAFETY EQUIVALENCY IN THE CANISTER STORAGE BUILDING AND COLD VACUUM DRYING FACILITY FINAL SAFETY ANALYSIS REPORTS

1.0 INTRODUCTION

This document identifies information that is to be included in the Canister Storage Building (CSB) and Cold Vacuum Drying Facility (CVDF) Final Safety Analysis Reports (FSAR) in addition to the information required by U.S. Department of Energy (DOE) Order 5480.23, *Nuclear Safety Analysis Reports*, and DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*. For the CSB, this additional information, unique to spent nuclear fuel storage, is that required by Title 10, *Code of Federal Regulations*, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Section 72.24, "Contents of Application, Technical Information" (10 CFR 72), and by U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 3.48, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)* (NRC 1989). For the CVDF, this additional information, unique to the conditioning of the spent nuclear fuel, is that required by Regulatory Guide 3.26, *Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants* (NRC 1975). Deviations from the content specified in DOE Order 5480.23 or DOE-STD-3009-94 shall be evaluated for compliance with the content requirements of the applicable NRC regulatory guide.

2.0 BACKGROUND

In Memorandum EM-36-3.1.6.7, *Concurrence with the K-Basins Spent Nuclear Fuel Project Policy on Nuclear Safety Requirements* (Grumbly 1995), (hereafter referred to as the Policy), DOE established the requirement that new Spent Nuclear Fuel Project (SNFP) facilities achieve "nuclear safety equivalency" to comparable NRC-licensed facilities. For that Policy (Grumbly 1995), nuclear safety equivalency was defined as conformance with

- Technical requirements that meet the nuclear safety objectives of the NRC regulations for fuel treatment and storage facilities, including requirements regarding radiation exposure limits, safety analysis, design, and construction
- Administrative requirements that meet the objectives of the major elements of the NRC licensing process, including formally documented design and safety analyses, independent technical review, and opportunity for public involvement.

Technical requirements, in the context of the Policy (Grumbly 1995) and as interpreted by the Regulatory Requirements Team, are the design and

construction measures (as opposed to preoperational or operational measures) that are mandated by the NRC regulations. In addition, the Policy specifically excludes those requirements that address only environmental, Occupational Safety and Health Administration, chemical accident safety, and other non-nuclear safety issues (Grumbly 1995).

NRC guidance that provides acceptable means to implement the regulations was defined as optional rather than mandatory in the Policy (Grumbly 1995). However, as a prudent step in implementing the Policy's objectives, NRC guidance that may have direct relevance to SNFP Path Forward activities (e.g., NRC Regulatory Guide 3.60, *Design of an Independent Spent Fuel Storage Installation [Dry Storage]* [NRC 1987]) was reviewed and evaluated for incorporation of additional requirements believed warranted to achieve nuclear safety equivalency.

Implementation of the Policy (Grumbly 1995) for the CSB is documented in two reports:

- WHC-SD-SNF-DB-002, *Spent Nuclear Fuel Project Path Forward, Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities* (Garvin 1996b)
- WHC-SD-SNF-DB-003, *Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements* (Garvin 1996a).

The first report documents the detail of the equivalency review and identifies actions for consideration that may be necessary to demonstrate nuclear safety equivalency. The second report presents the SNFP's position on each action for consideration identified in the first report and transforms those identified for implementation into a requirements format.

Information included in WHC-SD-SNF-DB-002 (Garvin 1996b) that relates to implementation of the design and construction aspects of 10 CFR 72 was based substantially on *Comparison of Nuclear Regulatory Commission and Department of Energy Requirements for Independent Spent Nuclear Fuel Storage Facilities* (Sciencetech 1995), a report written by Sciencetech, Inc., under DOE contract DE-AC01-92EW50504.

WHC-SD-SNF-DB-003 (Garvin 1996a) contains 29 equivalency items, including the following as item 24:

"Include in the SNF Project Path Forward Integrated Safety Management Plan the requirement to provide for the CSB SAR the information called for in Section 72.24 and Regulatory Guide 3.48 that is not required in DOE 5480.23 and DOE-STD-3009-94 and that is unique to spent nuclear fuel storage."

"Include in the SNF Project Path Forward Integrated Safety Management Plan the requirement to provide for the CVDF and HCS SARs, in consideration of conditioning processes and safety features, relevant information called for in Regulatory Guide 3.26 that is not required in DOE 5480.23 and DOE-STD-3009-94 and that is unique to the conditioning processes. (The review to identify any additional applicable information for the CVD facility or HCS SARs should not occur until the associated processes and safety features are better defined.)"

3.0 IMPLEMENTATION

3.1 CANISTER STORAGE BUILDING

Tables 1 and 2 present the information required by NRC Regulatory Guide 3.48 (NRC 1989) and Section 72.24 (10 CFR 72) that is unique to the storage of spent nuclear fuel. The items are listed in Tables 1 and 2 without regard to whether DOE Order 5480.23 or DOE-STD-3009-94 require the inclusion of such information in the FSAR for a nonreactor nuclear facility. In developing these tables, the requirements of Regulatory Guide 3.48 (NRC 1989) were reviewed first as the Regulatory Guide is more detailed in terms of the format and content requirements of the FSAR than is Section 72.24 (10 CFR 72). Items required by Section 72.24 (10 CFR 72), but not by Regulatory Guide 3.48 (NRC 1989), were then identified and included in Table 2.

Table 3 lists the additional items, arranged in DOE-STD-3009-94 format, that must be added to the CSB FSAR. It includes those items identified in Tables 1 and 2 that are not required by DOE Order 5480.23 or by DOE-STD-3009-94 and, therefore, must be added to implement item 24 of WHC-SD-SNF-DB-003. Also included in Table 3 are items that will document implementation of the remaining 28 equivalency items. This latter set is identified in Table 3 by the reference to "Garvin 1996a."

Subsequent to the approval and concurrence of the NRC nuclear safety equivalence documents (Garvin 1996a and 1996b), the NRC issued for comment NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems* (NRC 1996). This document has been reviewed to gain insight into the content requirements of Regulatory Guide 3.48 (NRC 1989), and appropriate references have been included in the attached tables. However, the draft NUREG-1536 (NRC 1996) was not reviewed to identify additional requirements for the CSB FSAR beyond those of Regulatory Guide 3.48 (NRC 1989) as this is not a requirement for equivalency.

3.2 COLD VACUUM DRYING FACILITY

WHC-SD-SNF-DB-003 (Garvin 1996a) states the review of Regulatory Guide 3.26 (NRC 1975) for the CVDF should be based upon an understanding of the associated processes and safety systems. This understanding was obtained from WHC-SD-SNF-SAR-002, *Safety Analysis Report for the Cold Vacuum Drying Facility, Phase 1, Supporting Civil/Structural Construction* (WHC 1996), through which the following CVDF processes and safety features were identified.

- Heating, ventilating, and air conditioning

Two high-efficiency particulate air filtered fan exhaust systems are provided. The first system exhausts the process bays, change rooms, transfer corridor, and mechanical rooms through high-efficiency particulate air filters and a common stack. Each CVDF bay is served by a dedicated ventilation system (i.e., with its own fan and cleanup train). Stack radionuclide releases are monitored. This system is important to maintaining secondary confinement when the

cask lid is removed. The second system exhausts the process bay local hoods. This is one system (with redundant fans) that services all of the hoods.

- MCO process systems

Each bay has its own process equipment skids containing all of the MCO processing equipment. The processing equipment is capable of drawing the K Basin water contained in the MCO into a collection system, purging warm dry gas through the MCO, and pulling a vacuum on the MCO. The equipment also includes a hot and chilled water system that establishes and controls the temperature of the MCO during the drying process. Portions of the processing system serve as an extension of the primary containment provided by the MCO.

Other mechanical systems that are distributed to each bay to support the process include a chilled water system for ventilation and process cooling; gases and deionized water; and waste water, floor drain, and radioactive liquid waste collection.

Maintaining the integrity of the process systems is an element in preventing MCO hydrogen deflagration and detonation. The process system is designed to maintain an inert atmosphere in system components and within the MCO and to prevent air ingress in order to preclude hydrogen deflagration and detonation. During normal operating process interruptions, or during other events that cause interruption of the process, the system is designed to isolate the MCO and to provide an inert gas purge to the MCO to prevent hydrogen burn events.

- Liquid waste system

Contaminated liquid waste water collected from the MCO during processing is transferred to a collection tank and later pumped to the radioactive liquid waste receiver tank. Design and operation of the liquid waste system provide for criticality control. The criticality prevention strategy is to prevent transfer of fissile material from the MCO by filtering the MCO liquid waste when transferring MCO liquid effluent to the collection tank, then to provide monitoring downstream of the filter or in the collection tank to detect failure of the filter and accumulations in the tank. The collection tank is monitored for fissile content before transfer to the radioactive liquid waste tank. The radioactive liquid waste tank also is monitored periodically for accumulations.

The following elements of the CVDF features listed above could be considered unique to the cold vacuum drying system:

- The function of the process system to drain an MCO, purge and vacuum dry the contents of the MCO, and backfill an MCO with inert gas
- The function of the process system to serve as an extension of the primary confinement provided by the MCO

- System integrity to preclude hydrogen deflagration and detonation
- Criticality control of the contents of the liquid waste system.

While the CVDF heating, ventilating, and air conditioning systems are very important to controlling the exposure of the workers, public, and environment to radionuclides, their design and operation has no unique considerations for the fuel conditioning process.

The Regulatory Guide 3.26 (NRC 1975) review was focused on the SAR content requirements of the guide as they relate to the four aspects of the CVDF listed above.

Table 4 lists the information identified from this review that should be included in the CVDF SAR. After this table was completed, its content was compared with the content requirements of DOE-STD-3009-94 to identify the information to be included in the SAR that is not required by DOE-STD-3009-94. The results of this review are provided in Table 5. This table suggests where the additional material should be included based upon the format requirements of the DOE standard. Also included in Table 5 are items that will document resolution of the remaining equivalency items. This latter set is identified in Table 5 by the reference to "Garvin 1996a."

4.0 REFERENCES

- 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, *Code of Federal Regulations*, as amended.
- DOE Order 5480.23, *Nuclear Safety Analysis Reports*, DOE 5480.23, U.S. Department of Energy, Washington, D.C.
- DOE-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.
- Garvin, L. J., 1996a, *Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements*, WHC-SD-SNF-DB-003, Rev. 2, Westinghouse Hanford Company, Richland, Washington.
- Garvin, L. J., 1996b, *Spent Nuclear Fuel Project Path Forward, Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities*, WHC-SD-SNF-DB-002, Rev. 2, Westinghouse Hanford Company, Richland, Washington.
- Grumbly, T. P., 1995, *Concurrence with the K-Basins Spent Nuclear Fuel Program Policy on Nuclear Safety Requirements*, (Memorandum EM-36-3.1.6.7 to Manager, U.S. Department of Energy, Richland Operations Office, July 20, 1995), U.S. Department of Energy, Washington D.C.

- NRC, 1975, *Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants*, Regulatory Guide 3.26, U.S. Nuclear Regulatory Commission, Washington, D.C.
- NRC, 1987, *Design of an Independent Spent Fuel Storage Installation (Dry Storage)*, Regulatory Guide 3.60, U.S. Nuclear Regulatory Commission, Washington, D.C.
- NRC, 1989, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)*, Regulatory Guide 3.48, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, D.C.
- NRC, 1996, *Standard Review Plan for Dry Cask Storage Systems*, NUREG-1536, draft for comment, U.S. Nuclear Regulatory Commission, Washington, D.C.
- Sciencetech, 1995, *Comparison of Nuclear Regulatory Commission and Department of Energy Requirements for Independent Spent Nuclear Fuel Storage Facilities*, Sciencetech, Incorporated, Rockville, Maryland
- WHC, 1996, *Safety Analysis Report for the Cold Vacuum Drying Facility, Phase 1, Supporting Civil/Structural Construction*, WHC-SD-SNF-SAR-002, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

Table 1. Items in Regulatory Guide 3.48^a that are Unique
to the Storage of Spent Nuclear Fuel. (8 sheets)

Section and title from Regulatory Guide 3.48 ^a	Item unique to spent nuclear fuel storage	Section and title of location in Final Safety Analysis Report for the Canister Storage Building
Chapter 1, "Introduction and General Description of Installation"		
Section 1.1, "Introduction"	Discuss the reason or need for an ISFSI.	Section ES.1, "Facility Background and Mission," and Section 2.3, "Facility Overview"
Section 1.2, "General Description of Installation"	Briefly discuss the normal capacity of the installation; the type, form, quantities, and potential sources of the spent fuel to be stored; and the waste products generated during ISFSI operation.	Section ES.2, "Facility Overview," and Section 2.3, "Facility Overview"
Section 1.3, "General Systems Description"	Present a summary description of the storage mode and arrangement of the storage structures to be used, including pertinent background information. Include a brief description of the operating systems; fuel handling, decay heat removal, and other auxiliary systems; and the site-generated waste treatment system.	Section 2.3, "Facility Overview"
Chapter 2, "Site Characteristics"		
Section 2.1.2, "Site Description"	Clearly delineate the area to be considered as the controlled area boundary if its boundaries are not the same as the boundaries of the site.	Section 1.3.1, "Geography" (This relates to the implementation of NRC equivalence for "important to safety".)
Section 2.1.2.1, "Other Activities Within the Site Boundary"	For any activity conducted within the area of the ISFSI, but not related to the operation of the facility, identify the activities involved, the boundaries within which the applicant will control such activities, and any potential interaction of such activities and the operation of the ISFSI.	Section 1.7, "Nearby Facilities" (This information is currently required by DOE-STD-3009-94.)
Section 2.1.2.2, "Boundaries for Establishing Effluent Release Limits"	Discuss the access control that enables compliance with the requirements of 10 CFR 72.104.	Section 1.3.1, "Geography"
Section 2.1.3, "Population Distribution and Trends"	Specify the current resident population, as well as projected future population changes, within each area formed by the concentric circles and radial lines of a population sector map.	Section 1.3.2, "Demography" (The equivalency review specifically recommended that the design bases for new SNFP Path Forward facilities address probable use of Hanford Site lands in the future [Garvin 1996]).
Section 2.6.1, "Basic Geologic and Seismic Information"	Present the methods used to determine that the site meets the design criteria of 10 CFR 72.102.	As nuclear safety equivalency is not to be demonstrated for seismic design, no additional seismic information need be provided.

Table 1. Items in Regulatory Guide 3.48^a that are Unique to the Storage of Spent Nuclear Fuel. (8 sheets)

Section and title from Regulatory Guide 3.48 ^a	Item unique to spent nuclear fuel storage	Section and title of location in Final Safety Analysis Report for the Canister Storage Building
Chapter 3, "Principal Design Criteria"		
Section 3.1, "Purposes of Installation"	Describe in general terms the mode of storage; the installation and its functions, operation, and storage capacity; and the types of fuel or high-level waste to be stored.	Section E.2, "Facility Overview," and Section 2.3, "Facility Overview" (This is the same information as required by Regulatory Guide 3.48 ^a for Section 1.1 with the addition of the discussion of the mode of storage.)
Section 3.1.1, "Materials to be Stored"	Provide a detailed description of the physical, thermal, and radiological characteristics of the spent fuel or high-level radioactive wastes to be stored. Include spent fuel characteristics such as specific power, burnup, decay time, and heat generation rates and high-level radioactive waste characteristics such as curie content, waste form, and generation rates.	Section 2.3, "Facility Overview"
Section 3.1.2, "General Operating Functions"	Provide information related to the overall functioning of the installation as a storage operation. Include information on onsite-generated waste processing, waste packaging and storage areas, transportation, and utilities.	Section 2.5, "Process Description"
Section 3.2.1, "Tornado and Wind Loadings"	Specify the design parameters applicable to the design tornado. Describe the methods used to convert the tornado and wind loadings into forces on structures. Show that the failure of any structure not being designed for tornado loads will not affect the ability of other SSCs important to safety to perform their intended design functions. Provide the dimensions, mass, energy, velocity, and other parameters for the tornado-driven missiles. Analyze the potential effect of such missiles on SSCs important to safety.	To achieve MRC nuclear safety equivalency, the CSB must be designed to withstand the effects of tornados and tornado missiles (Garvin 1996). ^d As DOE does not require that reactor or nonreactor nuclear facilities be designed for tornados and tornado missiles, this information needs to be added to Section 1.4.1, "Meteorology," to discuss the parameters that define the design basis tornado and tornado missiles, and to Section 2.4, "Facility Structure," in terms of the design of the CSB to withstand the effects of the tornado and tornado missiles as required to protect safety class 1 SSCs.
Section 3.2.3, "Seismic Design"	Justify the use of response spectral shapes other than those in Regulatory Guide 1.60 ^e if they are proposed for design of the storage structures or other structures that are important to safety. Present the earthquake time functions or other data from which these response spectral shapes were derived.	As nuclear safety equivalency is not to be demonstrated for seismic design, no additional seismic information need be provided. ^f This position is documented in MRC-SD-SNF-DB-004.

Table 1. Items in Regulatory Guide 3.48^a that are Unique to the Storage of Spent Nuclear Fuel. (8 sheets)

Section and title from Regulatory Guide 3.48 ^a	Item unique to spent nuclear fuel storage	Section and title of location in Final Safety Analysis Report for the Canister Storage Building
Chapter 4, "Installation Design"		
Section 4.1.2.2, "Controlled Area"	Show the controlled area established by the criteria in 10 CFR 72.106. ^c	Section 1.3.1, "Geography"
Section 4.3.2.2, "Safety Consideration and Controls"	Itemize and discuss the mechanisms and the sequence and timing of events that will occur in the event of a partial loss of normal power and in the event of a total loss of normal power.	Section 3.3, "Hazard Analysis" and Section 3.4, "Accident Analysis"
Section 4.7, "Spent Fuel and High-Level Radioactive Waste Handling Operation Systems"	<p>Describe the functions and design bases for any systems and structures performing the following functions: receipt and inspection of loaded shipping casks, cask unloading, spent fuel or high-level radioactive waste transfer and examination, fuel assembly or disassembly, placement of spent fuel in a container, container sealing and testing, short-term storage of spent fuel or high-level radioactive waste container, decontamination of shipping cask, safety SSC and drywell loading and preparation for storage, safety SSC transfer to storage, removal of spent fuel or high-level radioactive waste container from storage site to shipping cask, and placement of damaged fuel element in a container. Include the following items:</p> <ul style="list-style-type: none"> • Analysis and design procedures for tornado, earthquake, fire, explosion, and differential subsidence effects • General analysis and design procedures for normal, off-normal, and special loadings and load combinations • Allowable foundation loads and deflections and deformation stresses for structures • Considerations given to combinations of stress loadings • Provisions and methods for making connections between planned structures and future modifications and additions. 	Chapter 4.0, "Safety Structures, Systems, and Components"

Table 1. Items in Regulatory Guide 3.48^a that are Unique to the Storage of Spent Nuclear Fuel. (8 sheets)

Section and title from Regulatory Guide 3.48 ^a	Item unique to spent nuclear fuel storage	Section and title of location in Final Safety Analysis Report for the Canister Storage Building
Section 4.7.3, "Individual Unit Description"	<p>List each operational unit sequentially from the receipt of spent fuel or high-level radioactive waste through the various operations. The following are typical items to be included: shipping cask receiving and inspection, cask unloading, spent fuel or high-level radioactive waste canister transfer, spent fuel or high-level radioactive waste storage, hot-cell operations, and control locations. Show the location of each by use of engineering drawings.</p> <p>Describe the function of the individual operational areas, and discuss the performance objectives.</p> <p>Describe the components in the area under discussion. Use individual equipment sketches, layouts of equipment location to identify aspects of the components that must be relied on, and limits imposed on the design to achieve safety objectives.</p> <p>Present the design codes used and additional specifications necessary to provide a sufficient margin of safety under normal and accident conditions to ensure that a single failure will not result in the release of significant radioactive material. Include detail on backup provisions and interfaces with other areas. Also include a discussion of the features used to ensure that operating personnel are protected from radiation and contamination and that criticality will not occur.</p>	<p>DOE-STD-3009-94^b does not provide specific guidance as to where descriptions of the facility operations should be provided. However, it seems appropriate that much of this information should be included in Section 2.5, "Process Description."</p> <p>DOE-STD-3009-94^b requires that design codes and standards that are pertinent to the safety analysis be listed in Section 3.3, "Hazard Analysis" and Section 4.3, "Requirements." This standard requires analyses as to how the functional requirements are met, including single failure protection. This is to be included in Section 4.3.X.4, "System Evaluation," for safety class SSCs and Section 4.4.X.4, "System Evaluation", for safety-significant SSCs.</p> <p>Features used to ensure that operating personnel are protected from radiation and contamination are to be addressed on Chapter 7.0, "Radiation Protection."</p> <p>Considerations for prevention of criticality are to be addressed in Chapter 6.0, "Prevention of Inadvertent Criticality."</p>

Table 1. Items in Regulatory Guide 3.48^a that are Unique to the Storage of Spent Nuclear Fuel. (8 sheets)

Section and title from Regulatory Guide 3.48 ^a	Item unique to spent nuclear fuel storage	Section and title of location in Final Safety Analysis Report for the Canister Storage Building
Chapter 5, "Operation Systems"		
Chapter 5, "Operation Systems"	Provide a detailed description of all operations, including systems, equipment, and instrumentation and their operating characteristics. Identify potentially hazardous operating systems. In the details presented, designate the provisions made for operational safety features to ensure against a hazard, including, but not be limited to, backup or standby features and any systems necessary for curtailing operations under normal and off-normal conditions, for maintaining the installation in a safe condition, and for secondary confinement.	DOE-STD-3009-94 ^b does not provide specific guidance as to where descriptions of facility operations should be provided. However, it seems appropriate that much of this information should be included in Section 2.5, "Process Description."
Section 5.1.1, "Narrative Description"	Describe the proposed spent fuel and high-level radioactive waste handling and passive storage operations and relate them to the equipment and associated controls. Include pertinent ancillary activities (i.e., preparation of reactants, offgas handling, volume reduction of wastes, and decontamination). Identify the interfaces between systems, and discuss the safety aspects of the interfaces.	DOE-STD-3009-94 ^b does not provide specific guidance as to where descriptions of facility operations should be provided. However, it seems appropriate that much of this information should be included in Section 2.5, "Process Description."
Section 5.1.3.1, "Criticality Prevention"	Provide a summary description of the principal design features, procedures, and special techniques used to preclude criticality in all portions of the installation.	DOE-STD-3009-94 ^b requires that considerations for prevention of criticality be addressed in Chapter 6.0, "Prevention of Inadvertent Criticality."
Section 5.2.1.1, "Functional Description"	Present a flow diagram and functional description of the spent fuel or high-level radioactive waste receiving, storage, and retrieval systems, including provisions for handling defective fuel assemblies or damaged high-level radioactive waste containers. Include drawings or references to drawings as needed.	DOE-STD-3009-94 ^b does not provide specific guidance as to where descriptions of facility operations should be provided. However, it seems appropriate that much of this information should be included in Section 2.5, "Process Description."
Section 5.2.2, "Spent Fuel or High-Level Radioactive Waste Storage"	Describe the operations used for transfer of spent fuel assemblies or high-level waste containers to the storage positions, the storage surveillance program, and the operations used for removal from the storage position. Describe all features, systems, and special techniques included in the system that provide for the safety of the operation under both normal and off-normal conditions. Include the limits selected for a commitment to action.	DOE-STD-3009-94 ^b does not provide specific guidance as to where descriptions of facility operations should be provided. However, it seems appropriate that much of this information should be included in Section 2.5, "Process Description."

Table 1. Items in Regulatory Guide 3.48^a that are Unique to the Storage of Spent Nuclear Fuel. (8 sheets)

Section and title from Regulatory Guide 3.48 ^a	Item unique to spent nuclear fuel storage	Section and title of location in Final Safety Analysis Report for the Canister Storage Building
<p>Chapter 6, "Site-Generated Waste Confinement and Management"</p> <p>This section addresses waste that is generated as a result of the ISFSI operation. It does not refer to the high-level radioactive wastes that may be stored in the facility.</p>		
Section 6.5, "Radiological Impact of Normal Operations - Summary"	Identify the locations beyond the restricted area, as defined in 10 CFR 20.3(g)(14), ^g and beyond the controlled area, as defined in 10 CFR 72.3, ^c that are potentially impacted by radioactive material and effluents.	Section 1.3.1, "Geography," and Section 3.3.2.3.4, "Environmental Protection"
<p>Chapter 7, "Radiation Protection"</p> <p>This chapter requires detailed discussion of the implementation of Regulatory Guide 8.8.^h As implementation of MRC nuclear safety equivalency requires the incorporation of the design requirements of Regulatory Guide 8.8^h into the design of new SNFP Path Forward facilities, the information required in Regulatory Guide 3.48^a regarding implementation of the design aspects of Regulatory Guide 8.8^h should be included in the CSB FSAR.</p>		
<p>Chapter 8, "Accident Analyses"</p> <p>This chapter is to provide a discussion of analyses for off-normal operations and accidents. Site characteristics that have a bearing on the safety analyses are also to be discussed.</p> <p>For the analyses of off-normal operations, Regulatory Guide 3.48^a refers to events of the first and second type as discussed in ANSI/ANS-57.9-1992¹ (ANSI/ANS-57.9-1992¹ uses the terminology "Design Event I, II, III and IV"). Events of the first type are defined in ANSI/ANS-57.9-1992, Section 2, "Definitions," as events that are expected to occur regularly or frequently in the course of normal operation of the ISFSI. Events of the second type are defined as those not occurring regularly, but expected to occur with moderate frequency or on the order of once during a calendar year of ISFSI operation. In practice, utilities applying for licenses under 10 CFR 72,^c have not addressed events of the first type in Chapter 8 of their safety analysis reports as off-normal conditions or accidents. Such events have no consequence as they are within the operating design requirements for the facility. For example, implementation of this aspect of ANSI/ANS-57.9-1992¹ would require analyses of normal activities related to transportation. For example, implementation of this aspect of ANSI/ANS-57.9-1992¹ would require analyses of normal activities related to transportation, spent fuel transfer, handling of radioactive waste, fuel preparation, and placement of the spent fuel in the storage locations. These would be addressed without any postulated equipment failure or operator error. Some safety analysis reports have simply provided a list of the design criteria for normal operating conditions. For the CSB FSAR, written to the format of DOE-STD-3009-94,^d the confirmation that SSCs are designed for the appropriate loadings can be discussed in Chapter 4.0, "Safety Structures, Systems, and Components." This suggestion is supported by NUREG-1536,^j which states that Chapter 11, "Accident Analysis," should address off-normal and accident conditions and that loading for normal operation should be addressed in Chapter 3, "Structural Evaluation."</p> <p>For the analyses of accidents, Regulatory Guide 3.48^a refers to events that would potentially result in a dose of 25 mrem or greater beyond the controlled area and to events of the third and fourth type as discussed in ANSI/ANS-57.9-1992¹ (ANSI/ANS-57.9-1992¹ uses the terminology "Design Event I, II, III and IV"). Events of the third type are defined as that set of infrequent events that could be reasonably expected to occur during the lifetime of the ISFSI. Events of the fourth type are postulated because their consequences may result in the maximum potential impact on the immediate environs. Their consideration establishes a conservative design basis for certain systems that are important to confinement.</p>		

Table 1. Items in Regulatory Guide 3.48^a that are Unique to the Storage of Spent Nuclear Fuel. (8 sheets)

Section and title from Regulatory Guide 3.48 ^a	Item unique to spent nuclear fuel storage	Section and title of location in Final Safety Analysis Report for the Canister Storage Building
<p>Chapter 8, "Accident Analyses" (continued)</p> <p>For the CSB FSAR, the requirement of DOE-STD-3009-94,^b Chapter 3.0, to perform a hazards analysis that considers the complete spectrum of accidents that may result from facility operations will implement the requirement of Regulatory Guide 3.48^a to consider events of the second, third, and fourth type, and events whose offsite dose could equal or exceed 25 mrem. Safety analysis reports written to meet the requirements of 10 CFR 72^c provide analyses for the failure of the spent nuclear fuel containment vessel even if no credible scenarios were identified that could result in a breach of the vessel (NUREG-1536,^d Section V). The radionuclide releases are limited to fission product gases. The NRC assumes release of 30% of the krypton and 10% of the ¹³⁵I (NUREG-1536^d, Chapter 7.0, Section V.2.). The reported offsite doses are typically less than 2 rem. For the CSB FSAR, the need to consider beyond design basis events will be addressed by the requirement of DOE-STD-3009-94,^b Section 3.4.3, "Beyond Design Basis Accidents," that accidents beyond the design basis be evaluated to provide a perspective for the residual risk associated with the operation of the facility. Thus, implementation of DOE-STD-3009-94^b will result in the selection of off-normal events and accidents similar to those required by Regulatory Guide 3.48^a. However, events designated as an ANSI/ANS-57.9-1992^e Design Event 1 that have no consequences to the facility worker, collocated worker, public, or environment will be addressed by confirmation that the operational design loadings for the CSB will not be exceeded during normal facility operation.</p> <p>Regulatory Guide 3.48^a does require the presentation of information regarding the analysis of off-normal conditions and accidents that is not specifically stated in DOE-STD-3009-94,^b Section 3.4, "Accident Analysis." However, these differences do not relate to spent nuclear fuel storage.</p> <p>Section 8.2 of Regulatory Guide 3.48^a requires that accident analyses address situations wherein direct radiation or radioactive materials may be released in such quantity as to endanger personnel within the controlled area. The Regulatory Guide, however, does not require that a radiological dose assessments be made for such events. This requirement is addressed by DOE-STD-3009-94,^b which requires that Section 3.3.2.3.3 of the SAR summarize the major features protecting workers from hazards of facility operation. It should be noted that NUREG-1536^d only requires dose assessment for normal and anticipated occurrences (not to exceed 25 mrem beyond the controlled area) and for design basis accidents for any individual located at or beyond the nearest controlled area boundary (not to exceed 5 rem).</p>		
<p>Chapter 9, "Conduct of Operations"</p> <p>This chapter is to provide information on the applicant's organizational structure, preoperational testing and facility operation, training programs, normal operations, emergency planning, decommissioning plan, physical security, and safeguards and contingency plans. Some of the information to be provided is unique to the storage of spent nuclear fuel, such as record keeping (9.4.2), emergency planning (9.5), and decommissioning (9.6). However, as this information is not unique to spent nuclear fuel storage, it does not need to be included in the CSB FSAR. Also, while the NRC requirements for physical security and safeguards could involve design and construction issues, the NRC nuclear equivalency review^f did not identify any such issues from either 10 CFR 72^c or 10 CFR 73.</p>		
<p>Chapter 10, "Operating Controls and Limits"</p> <p>This chapter is to provide information on the development of licensing conditions and technical specifications. As none of the required information is unique to spent nuclear fuel storage, no additional information is required in the CSB FSAR.</p>		

Table 1. Items in Regulatory Guide 3.48^a that are Unique to the Storage of Spent Nuclear Fuel. (8 sheets)

Section and title from Regulatory Guide 3.48 ^a	Item unique to spent nuclear fuel storage	Section and title of location in Final Safety Analysis Report for the Canister Storage Building
<p>Chapter 11, "Quality Assurance"</p> <p>This chapter is to include a discussion of the applicant's quality assurance program to ensure that it meets the requirements of 10 CFR 72, Subpart G. The NRC nuclear safety equivalency review established that the combination of 10 CFR 830.120^b and the WMC Quality Assurance Manual,^c of 1.0 are equivalent to NRC requirements.^d Therefore, there is no need for the CSB FSAR to address the specific requirements of 10 CFR 72, Subpart G.</p>		

- ^a NRC, 1989, Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage), Regulatory Guide 3.48, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, D.C.
- ^b DOE-STD-3009-94, 1994, Preparation Guide for U.S. Department of Energy Monoreactor Nuclear Facility Safety Analysis Reports, U.S. Department of Energy, Washington, D.C.
- ^c 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Code of Federal Regulations, as amended.
- ^d Garvin, L. J., 1996, Spent Nuclear Fuel Project Path Forward, Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities, WMC-SD-SNF-DB-003, Rev. 2, Westinghouse Hanford Company, Richland, Washington.
- ^e NRC, 1973, Design Response Spectra for Seismic Design of Nuclear Power Plants, Regulatory Guide 1.60, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, D.C.
- ^f Garvin, L. J., 1996, Spent Nuclear Fuel Project Seismic Design Criteria, NRC Equivalency Evaluation Report, WMC-SD-SNF-DB-004, Rev. 2, Westinghouse Hanford Company, Richland, Washington.
- ^g 10 CFR 20, "Standards for Protection Against Radiation," Code of Federal Regulations, as amended.
- ^h NRC, 1978, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable, Regulatory Guide 8.8, U.S. Nuclear Regulatory Commission, Washington, D.C.
- ⁱ ANSI/ANS, 1992, Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type), ANSI/ANS-57.9-1992, American Nuclear Society, La Grange Park, Illinois.
- ^j NRC, 1996, Standard Review Plan for Dry Cask Storage Systems, NUREG-1536, draft for comment, U.S. Nuclear Regulatory Commission, Washington, D.C.
- ^k Garvin, L. J., 1996, Spent Nuclear Fuel Project Path Forward - Additional Nuclear Regulatory Commission Requirements, WMC-SD-SNF-DB-002, Rev. 2, Westinghouse Hanford Company, Richland, Washington.
- ^l 10 CFR 73, "Physical Protection of Plants and Materials," Code of Federal Regulations, as amended.
- ^m 10 CFR 830.120, "Quality Assurance," Code of Federal Regulations, as amended.
- ⁿ WMC-CN-4-2, Quality Assurance Manual, Fluor Daniel Hanford, Incorporated, Richland, Washington.

CSB = Canister Storage Building.

DOE = U.S. Department of Energy.

FSAR = Final safety analysis report.

NRC = U.S. Nuclear Regulatory Commission.

ISFSI = Independent spent fuel storage installation.

SNFP = Spent Nuclear Fuel Project.

SSC = structure, system, and component.

Table 2. Items in Title 10, *Code of Federal Regulations*, Section 72.24,^a that are Unique to the Storage of Spent Nuclear Fuel. (2 sheets)

Section of Title 10, <i>Code of Federal Regulations</i> , Part 72 ^a	Item unique to spent nuclear fuel storage	Section and title of location in Final Safety Analysis Report for the Canister Storage Building
72.24(a)	If the proposed ISFSI is to be located on the site of a nuclear power plant or other licensed facility, evaluate potential interactions.	Section 1.7, "Nearby Facilities." This information is currently required by DOE-STD-3009-94. ^b The presence of the MCS Annex will be addressed in Section 2.5, "Process Systems," and Chapter 3.0, "Hazard and Accident Analyses," when the MCS Annex is added to the CSB SAR.
72.24(b)	Provide a description and discussion of the ISFSI structures with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.	Chapter 4.0, "Safety Structures, Systems, and Components." Also see response provided in Table 1 for Regulatory Guide 3.48, ^c Section 4.7.
72.24(c)(1)	Address the design criteria for the ISFSI or MRS pursuant to Subpart F of 10 CFR 72, ^d with identification and justification for any additions to or departures from the general design criteria.	Chapter 2.0, "Facility Description"
72.24(e)	Describe the means for controlling and limiting occupational radiation exposures within the limits of 10 CFR 20. ^e	Chapter 7.0, "Radiation Protection." The extent to which 10 CFR 20 ^d is to be implemented for the CSB is defined in MHC-SD-SNF-DB-003. ^f
72.24(1)	Address SSCs important to safety whose functional adequacy or reliability have not been demonstrated.	Chapter 4.0, "Safety Structures, Systems, and Components."
72.24(l)(1)	Provide an estimate of the quantity of each of the principal radionuclides expected to be released annually to the environment in liquid or gaseous effluents produced during normal operations.	Chapter 7.0, "Radiation Protection." This is currently not a requirement of DOE-STD-3009-94. ^b As there will be no normal radioactive liquid releases, the required information can be obtained from the air permitting activities.
72.24(m)	Perform calculations of individual dose equivalent or committed dose equivalent for direct exposure, inhalation, and ingestion occurring as a result of the postulated design basis event.	Chapter 8.0, "Hazard and Accident Analyses."

Table 2. Items in Title 10, *Code of Federal Regulations*, Section 72.24,^a that are Unique to the Storage of Spent Nuclear Fuel. (2 sheets)

Section of Title 10, <i>Code of Federal Regulations</i> , Part 72 ^a	Item unique to spent nuclear fuel storage	Section and title of location in Final Safety Analysis Report for the Canister Storage Building
72.24(n)	Provide a description of the quality assurance program that satisfies the requirements of Subpart G. ^b The description must identify SSCs important to safety.	While NRC requirements for quality assurance could involve design and construction issues, the NRC nuclear equivalency review ^c established that the combination of 10 CFR 830.120 ^d and the existing Quality Assurance Manual, ^e QR 1.0, are equivalent to NRC requirements. ^f Therefore, the SAR does not need to address the requirements of Subpart G. ^g SSCs that would be designated as important to safety for a 10 CFR 72 ^h licensed facility will be identified in Chapter 3.0, "Hazards and Accident Analyses."
72.24(o)	Describe the detailed security measures for physical protection, including design features and the plans required by Subpart H. ⁱ	While NRC requirements for physical security and safeguards could involve design and construction issues, the NRC nuclear equivalency review ^j did not identify any such issues from 10 CFR 72. ^k Therefore the SAR does not need to address the requirements of Subpart H.

^a10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," *Code of Federal Regulations*, as amended.

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

^cNRC, 1989, *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)*, Regulatory Guide 3.48, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, D.C.

^d10 CFR 20, "Standards for Protection Against Radiation," *Code of Federal Regulations*, as amended.

^eGarvin, L. J., 1996, *Spent Nuclear Fuel Project Path Forward, Nuclear Safety Equivalency to Comparable NRC-licensed Facilities*, WNC-SD-SNF-DB-002, Rev. 2, Westinghouse Hanford Company, Richland, Washington.

^f10 CFR 830.120, "Quality Assurance Requirements," *Code of Federal Regulations*, as amended.

^gWNC-CM-4-2, *Quality Assurance Manual*, Fluor Daniel Hanford, Incorporated, Richland, Washington.

CSB = Canister Storage Building.

HCS = Hot Conditioning System.

ISFSI = independent spent fuel storage installation.

MRS = monitored retrievable storage.

NRC = U.S. Nuclear Regulatory Commission.

SAR = safety analysis report.

SSC = structure, system, and component.

Table 3. Information to be Added to the Final Safety Analysis Report for the Canister Storage Building that is not Required by DOE Order 5480.23^a or by DOE-STD-3009-94^b but is Needed to Implement Nuclear Safety Equivalency. (8 sheets)

Section and title from DOE-STD-3009-94 ^b	Material to be added (source)	Comment
"Executive Summary"		
Section ES.1, "Facility Background and Mission"	Discuss the need for the CSB (Regulatory Guide 3.48, ^c Section 1.1).	
Section ES.2, "Facility Overview"	Briefly discuss the mode of storage; its functions and operation; the normal capacity of the CSB; the types, forms, quantities, and potential sources of the spent fuel and high-level waste to be stored; and the waste products generated during CSB operation (Regulatory Guide 3.48, ^c Sections 1.2 and 3.1).	It is not expected that any waste products will be generated.
Section ES.7, "Safety Analysis Report Organization"	Discuss the manner in which the FSAR has been augmented to add that information from Regulatory Guide 3.48 ^c and 10 CFR 72.24 ^d that is unique to the storage of spent nuclear fuel (Garvin 1996). ^e	
Chapter 1.0, "Site Characteristics"		
Section 1.3.1, "Geography"	<p>Clearly delineate the area to be considered as the controlled area boundary if its boundaries are not the same as the boundaries of the site (Regulatory Guide 3.48,^c Section 2.1.2). Discuss the access control (10 CFR 72.106(b))^d that enables compliance with the requirements of 10 CFR 72.104^d (Regulatory Guide 3.48,^c Section 2.1.2.2).^d Show the controlled area established by the criteria in 10 CFR 72.106^d (Regulatory Guide 3.48,^c Section 4.1.2.2). Address the locations beyond the restricted area, as defined in 10 CFR 20.3(g)(14), and beyond the controlled area, as defined in 10 CFR 72.3,^a that are potentially impacted by radioactive material and effluents (Regulatory Guide 3.48,^c Section 6.5).</p> <p>Discuss the probable uses of Hanford Site land in the future and how this has been considered in the design bases of the CSB (Regulatory Guide 3.48,^c Section 2.1.3).</p>	These relate to the implementation of NRC equivalency for "important to safety."
Section 1.4.1, "Meteorology"	<p>Specify the design parameters applicable to the design tornado. Provide the dimensions, mass, energy, velocity, and other parameters for the tornado-driven missiles (Regulatory Guide 3.48,^c Section 3.2.1).</p> <p>Discuss the incorporation of a design basis tornado (including translational velocity, rotational velocity and pressure differential) and tornado missile for safety-class SSCs into the design of the CSB, taking into consideration the most recent version of Regulatory Guide 1.76,^g SECY-93-087,^h and NUREG/CR-4461,ⁱ and potential revisions to Standard Review Plan 3.5.1.4,^j Revision 2. Refer to WMC-SD-SNF-DB-002, Table 5.c,^k 10 CFR 72.24,^d and 10 CFR 72.122^d for related information (Garvin 1996).^e</p>	DOE does not impose the requirement to design reactor or nonreactor nuclear facilities for tornadoes and tornado missiles. This is an NRC nuclear safety equivalency issue that needs to be addressed in the FSAR (WMC-SD-SNF-DB-003 ^l and WMC-SD-SNF-DB-009 ^l).

Table 3. Information to be Added to the Final Safety Analysis Report for the Canister Storage Building that is not Required by DOE Order 5480.23^a or by DOE-STD-3009-94^b but is Needed to Implement Nuclear Safety Equivalency. (8 sheets)

Section and title from DOE-STD-3009-94 ^b	Material to be added (source)	Comment
Section 1.7, "Nearby Facilities"	Identify the locations beyond the restricted area, as defined in 10 CFR 20.3(g)(14), ^a and beyond the controlled area, as defined in 10 CFR 72.3, ^a that are potentially impacted by radioactive material and effluents (Regulatory Guide 3.48, ^c Section 6.5).	
Chapter 2.0, "Facility Description"	<p>Address the compliance of the design and construction of the CSB to the general design criteria of 10 CFR 72,^a Subpart F. Include identification and justification for any additions or departures from the criteria (10 CFR 72.24(c)(1))^a.</p> <p>Discuss the margins of safety during normal operations and expected operational occurrences during the life of the CSB (10 CFR 72.24(d)(1))^a.</p>	
Section 2.4, "Facility Structure"	<p>Describe the methods used to convert the tornado and wind loadings discussed in Section 1.4.1, "Meteorology," into forces on structures. Show that the failure of any structure not being designed for tornado loads will not affect the ability of other SSCs important to safety to perform their intended design functions. Analyze the potential effect of tornado missiles on SSCs important to safety (Regulatory Guide 3.48,^c Section 3.2.1).</p> <p>Describe the ability of the facility to withstand the natural phenomena hazard hydrology design requirements discussed in 1.4.2, "Hydrology."^a Include a discussion of the static and dynamic consequences of all types of flooding (Regulatory Guide 3.48,^c Section 2.4.8). Discuss the consequences of exceeding the design basis for the snow and ice loading (credit can be taken for design margin) (Regulatory Guide 3.48,^c Section 2.4.2.3).</p> <p>If the CSB is located adjacent to another structure, provide assurance that any physical interaction between the structures would not impair the capability of either facility to perform its safety functions. (Note: Discuss the sharing of features other than structures in Section 2.5, "Process Description".) (Garvin 1996)^a</p>	See the comment provided for Section 1.4.1, "Meteorology," regarding the reason for adding discussions of tornadoes and tornado missiles.

Table 3. Information to be Added to the Final Safety Analysis Report for the Canister Storage Building that is not Required by DOE Order 5480.23^a or by DOE-STD-3009-94^b but is Needed to Implement Nuclear Safety Equivalency. (8 sheets)

Section and title from DOE-STD-3009-94 ^b	Material to be added (source)	Comment
Section 2.5, "Process Description"	<p>Provide a detailed description of all operations, including systems, equipment, and instrumentation and their operating characteristics. Identify potentially hazardous operating systems. In the details presented, designate the provisions made for operational safety features to ensure against a hazard, including, but not be limited to, backup and standby features and systems necessary for curtailing operations under normal and off-normal conditions, for maintaining the installation in a safe condition, and for secondary confinement (Regulatory Guide 3.48,^c Section 5.1).</p> <p>List each operational unit sequentially from the receipt of spent fuel or high-level radioactive waste through the various operations. The following are typical items to be included: shipping cask receiving and inspection, cask unloading, spent fuel or high-level radioactive waste canister transfer, spent fuel or high-level radioactive waste storage, hot-cell operations, and control locations. Show the location of each by use of engineering drawings.</p> <p>Describe the function of the individual operational areas, and discuss the performance objectives. Discuss the components in the area under discussion. Use individual equipment sketches, layouts of equipment location to identify aspects of the components that must be relied on, and limits imposed on the design to achieve safety objectives (Regulatory Guide 3.48,^c Section 4.7.3).</p> <p>Provide information related to the overall functioning of the installation as a storage operation. Include information on onsite-generated waste processing, waste packaging and storage areas, transportation, and utilities (Regulatory Guide 3.48,^c Section 3.1.2).</p> <p>Describe the proposed spent fuel and high-level radioactive waste handling and passive storage operations, and relate them to the equipment and associated controls. Discuss pertinent ancillary activities (i.e., preparation of reactants, offgas handling, volume reduction of wastes, and decontamination). In the description, identify the interfaces between systems, and discuss the safety aspects of the interfaces (Regulatory Guide 3.48,^c Section 5.1.1).</p> <p>Present a flow diagram and functional description of the spent fuel or high-level radioactive waste receiving, storage, and retrieval systems, including provisions for handling defective fuel assemblies or damaged high-level radioactive waste containers. Include drawings or references to drawings as needed (Regulatory Guide 3.48,^c Section 5.2.1.1).</p>	<p>Note that these paragraphs were taken from several locations within Regulatory Guide 3.48^c and, as such, some of the material is repetitive. The requirement is to provide the information requested, but not necessarily in the order presented here.</p> <p>It is important that all operations and processes that are identified as important to safety be addressed.</p>

Table 3. Information to be Added to the Final Safety Analysis Report for the Canister Storage Building that is not Required by DOE Order 5480.23^a or by DOE-STD-3009-94^b but is Needed to Implement Nuclear Safety Equivalency. (8 sheets)

Section and title from DOE-STD-3009-94 ^b	Material to be added (source)	Comment
Section 2.5, "Process Description" (continued)	<p>Describe the operations used for transfer of spent fuel assemblies or high-level waste containers to the storage positions; the storage surveillance program; and the operations used for removal from the storage position. Describe all features, systems, and special techniques included in the system that provide for the safety of the operation under both normal and off-normal conditions. Include the limits selected for a commitment to action (Regulatory Guide 3.48,^c Section 5.2.2).</p> <p>Provide assurance that sharing of common utilities and services and physical interaction between the CSB and other facilities do not impair the capability of either facility to perform its safety functions (Note: The consideration for sharing of structures is to be discussed in Section 2.4, "Facility Structure.") (Garvin 1996)^e</p>	
Chapter 3.0, "Hazards and Accident Analyses"	Evaluate those nearby activities identified in Section 1.6, "External Human-Generated Threats," that may represent a threat to the facility.	
Chapter 4.0, "Safety Structures, Systems, and Components"	<p>Describe the functions and design bases for any systems and structures performing the following functions: receipt and inspection of loaded shipping casks, cask unloading, spent fuel or high-level radioactive waste transfer and examination, fuel assembly or disassembly, placement of spent fuel in a container, container sealing and testing, short-term storage of spent fuel or high-level radioactive waste container, decontamination of shipping cask, safety SSC and drywell loading and preparation for storage, safety SSC transfer to storage, removal of spent fuel or high-level radioactive waste container from storage site to shipping cask, and placement of a damaged fuel element in a container. Include the following items (Regulatory Guide 3.48,^c Section 4.7 unless noted otherwise):</p> <ul style="list-style-type: none"> • Analysis and design procedures for tornado, earthquake, fire, explosion, and differential subsidence effects • General analysis and design procedures for normal, off-normal, and special loadings and load combinations • Allowable foundation loads and deflections and deformation stresses for structures • Considerations given to combinations of stress loadings • Provisions and methods for making connections between planned structures and future modifications and additions. • Unusual or novel design features (10 CFR 72.24(b))^d • SSCs whose functional adequacy or reliability have not been demonstrated (10 CFR 72.24(i))^d 	Chapter 2.0, "Facility Description," may be a more appropriate location for this information.

Table 3. Information to be Added to the Final Safety Analysis Report for the Canister Storage Building that is not Required by DOE Order 5480.23^a or by DOE-STD-3009-94^b but is Needed to Implement Nuclear Safety Equivalency. (8 sheets)

Section and title from DOE-STD-3009-94 ^b	Material to be added (source)	Comment
Section 4.3, "Safety-Class Systems, Structures, and Components"	<p>Demonstrate implementation of the NRC nuclear safety equivalency requirements documented in WNC-SD-SNF-DB-003.^c</p> <ul style="list-style-type: none"> • Document compliance to ANSI/ANS-57.9-1992.^m • If the CSB includes safety-class electrical and/or instrumentation and control systems, commit to using IEEE Standard 603-1991.ⁿ • If the safety-class electrical or instrumentation and control systems also use safety class lead storage batteries, commit to using IEEE Standard 484-1897^o and IEEE Standard 535-1986.^p • If the CSB includes safety-class electrical equipment that may be exposed to harsh environments during off-normal or postulated accident conditions, commit to the additional environmental qualification requirements identified in WNC-SD-SNF-DB-003.^q • If the CSB uses safety-class instrument air, motor-operated valves, or an open-cycle cooling system, document the use of NRC Generic Letters 88-14,^r 89-10 (with supplements),^s and 89-13^t as appropriate. 	
Section 4.4, "Safety-Significant Structures, Systems, and Components"	<p>State that the requirements of ANSI/ASME N509-1989^u and ANSI/ASME N510-1989^v have been incorporated into the design of safety-significant heating, ventilating, and air conditioning systems used to achieve onsite radiological limits or to implement the principals of ALARA (Garvin 1996).</p> <p>State that the CSB design has incorporated the design requirements of ANSI/ANS-57.1-1992^w and ANSI/ANS-57.2-1983.^x (Should any of this equipment be safety class, then move the discussion to Section 4.3, "Safety-Class Systems, Structures, and Components." Note that DOE Order 6430.1A^y requires, in Section 1320-S.2 for Irradiated Fissile Material Storage Facilities, that, "Special design features shall be considered to ensure safe loading, removal, and handling of IFM. These systems and equipment shall be designed as safety class items and shall protect against the dropping of shipping casks, IFM assemblies, and other items on the stored IFM" (Garvin 1996).^z</p>	

Table 3. Information to be Added to the Final Safety Analysis Report for the Canister Storage Building that is not Required by DOE Order 5480.23^a or by DOE-STD-3009-94^b but is Needed to Implement Nuclear Safety Equivalency. (8 sheets)

Section and title from DOE-STD-3009-94 ^b	Material to be added (source)	Comment
Chapter 6.0, "Prevention of Inadvertent Criticality"	<p>Include a commitment to a criticality safety k_{eff} value of 0.95 (Garvin 1996).^c Discuss the extent to which the requirements of ANSI/ANS-8.3-1986^d have been incorporated into the design.</p> <p>Note: Evaluate the multiccanister overpack design, which includes the basket configuration, and the quantities and form of the K Basin fuel as a basis for not incorporating criticality accident alarm systems in the CSB. This should be based on demonstrating through the safety analyses documented in the FSAR that criticality is not possible. As the design progresses, reconfirm the evaluation results. Refer to WMC-SD-SNF-DB-002, Table 6.a,^k and Regulatory Guide 8.12^z for additional information (Garvin 1996).^e</p>	
Chapter 7.0, "Radiation Protection"	<p>State that the CSB has incorporated control devices for access to high-radiation areas that conform to the requirements of 10 CFR 20.1601.¹ 10 CFR 20.1601¹ requires control devices for all high-radiation areas that meet the definition in 10 CFR 20.1003¹ of 0.1 rem in 1 hour at 30 cm, whereas 10 CFR 835^{aa} does not require incorporation of control devices until the dose rate in an accessible area reaches 1 rem in 1 hour at 30 cm. Control devices are hardware features, such as alarms or locked entryways, as opposed to administrative controls. Also state that the CSB has incorporated into the design for any unrestricted areas the 10 CFR 20.1301¹ hourly dose limit of 0.002 rem to the public from external sources during normal operations and anticipated occurrences (Garvin 1996).^e</p> <p>State that the CSB design and safety analysis apply the radiological exposure criteria of 10 CFR 72.104⁴ to any real individual of the public during normal operations and anticipated occurrences (Garvin 1996).^e These annual dose equivalent criteria are 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other critical organ; existing DOE requirements are 25 mrem to the whole body and 75 mrem to any critical organ.^e</p> <p>Provide an estimate of the principal gaseous radionuclides expected to be released annually and state that there will be no liquid release during normal operation.</p>	This also responds to 10 CFR 72.24(e) ² of Table 2.
Section 7.4, "ALARA Policy and Program"	Show that the design requirements of Regulatory Guide 8.8 ^{bb} have been implemented. Provide the information required by Regulatory Guide 3.48 ² for FSAR Section 7.3, "Radiation Protection Design Features," to show that the design requirements of Regulatory Guide 8.8 ^{bb} have been satisfied (Regulatory Guide 3.48, ² Section 7.3).	
Section 7.7, "Radiological Monitoring"	Document that the provisions for radiological monitoring will meet the requirements of 10 CFR 20, ¹ 10 CFR 70.59, ^{cc} and 10 CFR 835 ^{aa} (Garvin 1996). ^e	

Table 3. Information to be Added to the Final Safety Analysis Report for the Canister Storage Building that is not Required by DOE Order 5480.23^a or by DOE-STD-3009-94^b but is Needed to Implement Nuclear Safety Equivalency. (8 sheets)

Section and title from DOE-STD-3009-94 ^b	Material to be added (source)	Comment
Chapter 8.0, "Hazards and Accident Analysis"	Address the ingestion pathway.	
Section 11.4, "Fire Protection"	According to WNC-SD-SNF-DB-002, Table 5.b, Section 50.48, and Appendix R, the final design of the CSB was to be reevaluated to reconfirm that DOE Orders 5480.7A ^{dd} and 6430.1A ^{xx} provide adequate fire protection requirements to achieve nuclear safety equivalence. Aspects of the design to be reconfirmed were the use of a passive cooling system for multicatcher overpack cooling in the CSB and the lack of safety-class prevention or mitigation systems in the CSB. Further, fire protection requirements considered for incorporation into the design of the CSB were to take into account the implementation of 10 CFR 72.122(c) ⁵ to date for licensed Independent Spent Fuel Storage Installations (Garvin 1996). ⁵ Address the review and the additional requirements in the FSAR if it is determined that the requirements of DOE Orders 5480.7A ^{dd} and 6430.1A ^{xx} need to be augmented by the above-discussed NRC requirements.	
Chapter 13.0, "Human Factors"	According to WNC-SD-SNF-DB-003, ⁶ the CSB project was to review the NRC guidance of NUREG-0700 ^{cc} and Standard Review Plan 18.1 ^{ff} against DOE Order 6430.1A, Section 1300-12.4, to identify appropriate additional NRC guidance for design of the facility. The review was to consider the differences in complexity between a power reactor control room and that of the CSB. In the FSAR, discuss the results of this review and the extent to which the requirements of NUREG-0700 ^{cc} and Standard Review Plan 18.1 ^{ff} have been incorporated into the design of the CSB (Garvin 1996). ⁶	

^aDOE Order 5480.23, 1992, Nuclear Safety Analysis Reports, DOE 5480.23, U.S. Department of Energy, Washington, D.C.

^bDOE-STD-3009-94, 1994, Preparation Guide for U.S. Department of Energy Monoreactor Nuclear Facility Safety Analysis Reports, U.S. Department of Energy, Washington, D.C.

^cNRC, 1989, Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage), Regulatory Guide 3.48, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, D.C.

^d10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Code of Federal Regulations, as amended.

^eGarvin, L. J., 1996, Spent Nuclear Fuel Project Path Forward, Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities, WNC-SD-SNF-DB-003, Rev. 2, Westinghouse Hanford Company, Richland, Washington.

^f10 CFR 20, "Standards for Protection Against Radiation," Code of Federal Regulations, as amended.

^gNRC, 1974, Design Basis Tornado for Nuclear Power Plants, Regulatory Guide 1.76, U.S. Nuclear Regulatory Commission, Washington, D.C.

^hNRC, 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.

ⁱRamsdell, J. V., and G. L. Andrews, 1986, Tornado Climatology of the Contiguous United States, NUREG/CR-4461, prepared for the U.S. Nuclear Regulatory Commission by Pacific Northwest Laboratory, Richland, Washington.

^jNRC, 1981, Standard Review Plan, Section 3.5.1.4, "Missiles Generated by Natural Phenomena," Rev. 2, U.S. Nuclear Regulatory Commission, Washington, D.C.

^kGarvin, L. J., 1996, Spent Nuclear Fuel Project Path Forward - Additional Nuclear Regulatory Commission Requirements, WNC-SD-SNF-DB-002, Rev. 2, Westinghouse Hanford Company, Richland, Washington.

Table 3. Information to be Added to the Final Safety Analysis Report for the Canister Storage Building that is not Required by DOE Order 5480.23^a or by DOE-STD-3009-94^b but is Needed to Implement Nuclear Safety Equivalency. (8 sheets)

Section and title from DOE-STD-3009-94 ^b	Material to be added (source)	Comment
	¹ Tallman, A. M., 1996, <u>Canister Storage Building Natural Phenomena Hazards</u> , WNC-SO-SNF-DB-009, Rev. 4, Westinghouse Hanford Company, Richland, Washington.	
	² ANSI/ANS, 1992, <u>Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)</u> , ANSI/ANS-57.9-1992, American Nuclear Society, La Grange Park, Illinois.	
	³ IEEE, 1991, <u>IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations</u> , IEEE Std 603-1991, Institute of Electrical and Electronics Engineers, New York, New York.	
	⁴ IEEE, 1987, <u>IEEE Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations</u> , IEEE Std 484-1987, Institute of Electrical and Electronics Engineers, New York, New York.	
	⁵ IEEE, 1986, <u>IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations</u> , IEEE Std 535-1986, Institute of Electrical and Electronics Engineers, New York, New York.	
	⁶ NRC, 1988, <u>Instrument Air Supply System Problems Affecting Safety-Related Equipment</u> , Generic Letter 88-14, U.S. Nuclear Regulatory Commission, Washington, D.C.	
	⁷ NRC, 1989, <u>Safety-Related (1) Motor-Operated Valve Testing and Surveillance</u> , Generic Letter 89-10, U.S. Nuclear Regulatory Commission, Washington, D.C.	
	⁸ NRC, 1989, <u>Service Water System Problems Affecting Safety-Related Equipment</u> , Generic Letter 89-13, U.S. Nuclear Regulatory Commission, Washington, D.C.	
	⁹ ANSI/ASME, 1989, <u>Nuclear Power Plant Air-Cleaning Units and Components</u> , ANSI/ASME N509-1989, American Society of Mechanical Engineers, New York, New York.	
	¹⁰ ANSI/ASME, 1989, <u>Testing of Nuclear Air Treatment Systems</u> , ANSI/ASME N510-1989, American Society of Mechanical Engineers, New York, New York.	
	¹¹ ANSI/ANS, 1992, <u>Design Requirements for Light Water Reactor Fuel Handling System</u> , ANSI/ANS-57.1-1992, American Nuclear Society, La Grange Park, Illinois.	
	¹² ANSI/ANS, 1983, <u>American National Standard Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants</u> , ANSI/ANS-57.2-1983, American Nuclear Society, La Grange Park, Illinois.	
	¹³ DOE Order 6430.1A, 1989, <u>General Design Criteria</u> , U.S. Department of Energy, Washington, D.C.	
	¹⁴ ANSI/ANS, 1986, <u>Criticality Accident Alarm System</u> , ANSI/ANS-8.3-1986, American Nuclear Society, La Grange Park, Illinois.	
	¹⁵ NRC, 1988, <u>Criticality Accident Alarm Systems</u> , Regulatory Guide 8.12, U.S. Nuclear Regulatory Commission, Washington, D.C.	
	¹⁶ 10 CFR 835, "Occupational Radiation Protection," Code of Federal Regulations, as amended.	
	¹⁷ NRC, 1978, <u>Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable</u> , Regulatory Guide 8.8, Rev. 3, U.S. Nuclear Regulatory Commission, Washington, D.C.	
	¹⁸ 10 CFR 70, "Domestic Licensing of Special Nuclear Material," Code of Federal Regulations, as amended.	
	¹⁹ DOE Order 5480.7A, 1993, <u>Fire Protection</u> , U.S. Department of Energy, Washington, D.C.	
	²⁰ NRC, 1995, <u>Human System Interface Design Review Guidelines</u> , NUREG-0700, Rev. 1 (for comment), U.S. Nuclear Regulatory Commission, Washington, D.C.	
	²¹ NRC, 1984, <u>Standard Review Plan</u> , Section 18.1, "Human Factors Engineering, Control Room," NUREG-0800, U.S. Nuclear Regulatory Commission, Washington, D.C.	

ALARA = as low as reasonably achievable.

CSB = Canister Storage Building.

DOE = U.S. Department of Energy.

FSAR = final safety analysis report.

IFM = irradiated fissile material.

NRC = U.S. Nuclear Regulatory Commission.

SNFP = Spent Nuclear Fuel Project.

SSC = structure, system, and component.

Table 4. Items of Regulatory Guide 3.26^a that are Unique to Fuel Conditioning. (6 sheets)

Section and title from Regulatory Guide 3.26 ^a	Item unique to fuel conditioning	Cold Vacuum Drying Facility Safety Analysis Report section and title
1.1 Introduction	Describe the type and exposure of the fuel to be conditioned.	Executive Summary
1.3 General Process Description	Include a summary description of the process to be used, including process origin and background. Identify basic material flow and balance for the product and waste streams. Provide sufficient detail in the discussion and accompanying charts and tables to provide an understanding of the processes involved.	Section 2.5, Process Description
4.1 Purpose of Plant	Describe in general terms the plant, its functions and operation, process capacity, type of feed, and products.	Section 2.5, Process Description
4.1.1 Plant Feed	Provide a detailed description of the physical, chemical, and radiological characteristics of the spent fuel to be processed in the plant. Include feed specifications, such as fissile material limits, forms of the material, and packaging. Presentation of radioactivity characteristics should include irradiation history, minimum cooling time at receipt, time to processing, and fission product concentrations.	Section 2.5, Process Description
4.1.2 Plant Products and Byproducts	Identify the products and byproducts that will result from plant operation. Include product characteristics and specifications.	Section 2.5, Process Description
4.1.3 Facility Functions	Provide information related to the overall functioning of the plant as a reprocessing operation. Include onsite waste processing, waste disposal or holding areas, transportation, and utility and water supplies.	Section 2.5, Process Description
4.3.2.1 Confinement Barriers and Systems	<p>Discuss each method of confinement that will be used to ensure that there will be no uncontrolled release of radioactivity to the environment. Include for each method</p> <ol style="list-style-type: none"> 1. Criteria for protection against any postulated internal accident or external natural phenomena 2. Design criteria for vessels, piping, effluent systems, and backup confinement 3. Delineation of the extent to which each design is based on achieving the lowest practical level of releases from the operation of the plant. <p>Where the limits selected are consistent with proven practice, a referenced statement to that effect will suffice; where the limits extend beyond present practice, an evaluation and an explanation based upon developmental work and/or analysis should be provided. Those criteria may be expressed as explicit numbers or as general conditions.</p>	<p>Section 2.6, Confinement System</p> <p>These requirements are more specific than those required of DOE-STD-3009-94^b for Section 2.6.</p>
4.3.8 Industrial and Chemical Safety	Any specific design criteria should be described that are important to personnel and plant safety. Effects of various industrial accidents should be presented (e.g., fire, explosion), as well as hazardous chemical reactions (e.g., vigorous chemical reactions, hydrogen explosions).	The identification of specific design criteria is required by DOE-STD-3009-94 ^b for Section 8.2. The evaluation of the effects of various hazards is required by DOE-STD-3009-94 ^b for Section 3.3.2.

Table 4. Items of Regulatory Guide 3.26^a that are Unique to Fuel Conditioning. (6 sheets)

Section and title from Regulatory Guide 3.26 ^a	Item unique to fuel conditioning	Cold Vacuum Drying Facility Safety Analysis Report section and title
4.5 Decommissioning	As required by 10 CFR Part 50, ^c Appendix F, "Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities," arrangements must be included for safe decommissioning of the plant. Provide in this section the design criteria incorporated in the plant for fulfilling this requirement. Discuss the design philosophy to be used to allow ease of facility decontamination and future isolation of all radioactive materials from the public while maximizing the land area returned to the public domain.	These requirements are similar to those required of DOE-STD-3009-94 ^d for Chapter 16, "Provisions for Decontamination and Decommissioning."
5.4.11 Cold Chemical Systems	Where hazardous chemicals or materials are involved, discuss the provisions that will be made to mitigate accidents. Itemize the chemicals and materials that will be used, their quantities, and where they will be used, and codify with respect to hazard.	Provisions to mitigate hazardous materials accidents are required by DOE-STD-3009-94 ^d for Section 3.3.2 when the hazard is identified as having an unacceptable risk. The standard requires that Section 8.6 provide a discussion of hazard elimination and control for other risks not evaluated as specific accidents. Section 8.6 must also provide a description of the process for the identification and evaluation of hazards.
5.5 Items Requiring Further Development	<p>Identify, describe, and discuss those safety features or components for which further technical information will be required in support of the issuance of a construction permit but that has not been supplied in the PSAR. This section should:</p> <ul style="list-style-type: none"> Identify and distinguish between those technical information development programs that will be required to determine the adequacy of a new design and those that will be used to demonstrate the margin of conservatism of a proven design Characterize the specific technical information that must be obtained to demonstrate acceptable resolution of the problems Outline the program in sufficient detail to show how the information will be obtained Provide a schedule of completion of the program as related to the projected startup date of the proposed plant Discuss the design alternatives or operational restrictions available in the event that the results of the program do not demonstrate acceptable resolution of the problems. <p>Provide reasonable assurance that the alternatives that will be considered will be acceptable substitutes.</p>	These requirements are similar to those that would be required of DOE-STD-3009-94 ^d for Section 3.3.2.3.1 for a new facility.
6.1.1 Narrative Description (of Process Systems)	Describe the proposed process and relate it to the equipment and associated controls. Include in this discussion ancillary activities as pertinent to the use of the main process (i.e., preparation of reactants, offgas handling, volume reduction of wastes, and decontamination). In the description, identify the interfaces between systems and discuss the safety aspects of the interfaces.	Section 2.5, Process Description

Table 4. Items of Regulatory Guide 3.26^a that are Unique to Fuel Conditioning. (6 sheets)

Section and title from Regulatory Guide 3.26 ^a	Item unique to fuel conditioning	Cold Vacuum Drying Facility Safety Analysis Report section and title
6.1.2 Flowsheets (for Process Systems)	In support of the description above, supply flowsheets showing the process, materials and heat balances, and instrumentation. Provide identification of the process and effluent streams in sufficient detail that an independent review can be made to ensure a safe operation. Include stream flow quantities, activities, compositions, properties, sample points, and identification of primary control points. Provide the flow input characteristics for effluent control equipment, as well as its output, to show the efficiencies obtained. Sufficient detail should be given to provide source terms for radiation exposure determinations. Include equipment descriptions with dimensions, design and operating temperatures and pressures, materials of construction, special design features, and process limitations. Also provide appropriate engineering and process instrumentation details and flow diagrams.	Section 2.5, Process Description
6.1.3 Identification of Items for Safety Analysis Concern	Provide identification of areas or items for safety analysis concern. Reference this part of the chapter, as applicable, in subsequent discussion of design and operating features and items discussed in Chapter 2.0, "Summary Safety Analysis."	These requirements are similar to those that would be required of DOE-STD-3009-94 ^b for Section 3.3.2.3.1 for a new facility.
6.1.3.2 Chemical Safety	Provide a summary description of the principal chemical hazards (hydrogen, red oil, solvent flammability, fluorine) and the approaches used to preclude associated accidents.	This requirement is similar to that required of DOE-STD-3009-94 ^b for Chapter 8, "Hazardous and Material Protection."
6.1.3.3 Process Shutdown Modes	Describe the general plant conditions and surveillance needs in various shutdown modes--extended, short-term, emergency. Indicate the time required to shut down and start up for each mode.	Section 2.5, Process Description
6.1.3.4 Instrumentation	Provide a summary description of the instruments used to detect process conditions and the systems used to control the process. The description should include testability, redundancy, and failure conditions. Also, describe effluent and process monitors and data logger.	Section 2.5, Process Description
6.1.3.5 Remote and Contact Maintenance Techniques	Discuss the rationale and outline the techniques to be used for remote and contact maintenance. This should include a statement of the cells where the techniques apply. Include system and component spares.	
6.3.1 Fuel Receiving, Storage, Handling, and Transfer	Describe the systems associated with fuel receipt. From the design criteria, present the provisions for cooling, maintaining fuel assemblies in subcritical form, and shielding.	Section 2.5, Process Description DOE-STD-3009-94 ^b requires that Section 6.4 provide a discussion of the controls required to prevent criticality. The standard requires that Chapter 8 address radiological protection but does not specifically require a discussion of the provisions provided for shielding.

Table 4. Items of Regulatory Guide 3.26^a that are Unique to Fuel Conditioning. (6 sheets)

Section and title from Regulatory Guide 3.26 ^a	Item unique to fuel conditioning	Cold Vacuum Drying Facility Safety Analysis Report section and title
6.4.1.3 Design Description	Discuss the design bases, including materials of construction; pressure and temperature limits; detailed dimensions, especially as related to criticality consideration if not discussed elsewhere; corrosion allowances; and standards or codes used. Itemize material and fabrication specifications pertaining to the system in sufficient detail to relate to the Quality Assurance Program. Include such items as material of construction, identification, assembly, welding, installation, and testing.	Section 2.5, Process Description
6.4.1.4 Safety Criteria and Assurance	From the parameters discussed in the preceding subsections, summarize the criteria, the means of ensuring a safe system as constructed, operated, and maintained. Summarize those limits selected for commitment to action. Also, identify those items that can be characterized as being engineered safety features and that are considered additionally necessary beyond normal process operation and control. Emphasis should be placed on criticality considerations.	Section 2.5, Process Description
6.4.1.5 Operating Limits	Identify limits, conditions, and performance requirements in sufficient detail to make possible an evaluation as to whether a technical specification may be necessary. The interface relationship to other systems should be clearly described.	Section 2.5, Process Description
6.4.2 Component/Equipment Spares	Describe, in detail, design features that include installation of spare or alternative equipment to provide continuity of safety under normal and abnormal conditions. This may include vessels, jets, pumps, remote jumper pipes, heaters or coolers, and valves. Particular emphasis is needed on design selection to minimize radiation exposure for maintenance operations. Also, describe the bases for inspection, preventive maintenance, and testing programs to ensure continued safe functioning.	Section 2.5, Process Description
6.5 Process Support Systems	Although effluent handling systems may be considered process support, provision is made to discuss these elsewhere. To the extent a system is designed for complete recycle within the main process, it should be discussed in this section. Describe any cold chemical systems that will supply or regenerate reactants being used. Principal auxiliary backup equipment is to be discussed elsewhere.	Section 2.5, Process Description

Table 4. Items of Regulatory Guide 3.26^a that are Unique to Fuel Conditioning. (6 sheets)

Section and title from Regulatory Guide 3.26 ^a	Item unique to fuel conditioning	Cold Vacuum Drying Facility Safety Analysis Report section and title
6.5.1 Instrumentation and Control Systems	By reference to instrumentation engineering flowsheets and the process, discuss the instrumentation and control features associated with process control, process monitors, and alarms, and the relationship of one to the other. Identify aspects that must be relied upon to establish that adequate reliability is provided and that provisions have been included in the design to ensure continued safe operation or safe shutdown under accident conditions. Relate these to the design criteria presented elsewhere. Discuss how instrumentation and control systems monitor safety-related variables and operating systems over anticipated ranges for normal operation, for abnormal operation, for accident conditions, and for safe shutdown. Describe the redundancy of engineered safety features required to ensure adequate safety of process and utility operations. The safety-related variables and systems that require constant surveillance and control include the overall confinement system, each confinement barrier and its associated systems, and other process systems that affect the overall safety of the plant. Discuss the provisions for in situ testability of the instrumentation and control systems. Describe how instrumentation and control systems are designed to be fail-safe or to assume a state demonstrated to be acceptable if conditions such as disconnection, loss of energy or motive power, or adverse environments are experienced.	Section 2.5, Process Description
6.5.2 System and Component Spares	Describe in detail installation of spare or alternative instrumentation designed to provide continuity of operation under normal and abnormal conditions. Also, describe the bases for inspection, preventive maintenance, and testing programs to ensure continued safe functioning.	Section 2.5, Process Description
6.6 Control Room	Discuss how a control room and/or control areas are to be designed to permit occupancy and actions to be taken to operate the plant safely under normal conditions and to maintain the plant in a safe condition under accident or other abnormal conditions. Describe the redundancy that allows the plant to be put into a safe condition if any control room or control area is removed from service.	Section 2.5, Process Description
6.7 Sampling--Analytical	Discuss provisions for obtaining samples for process analysis and control necessary to ensure that operations are within prescribed limits. Describe the facilities and analytical equipment that will be available to perform the analyses, as well as the destination of laboratory wastes.	Section 2.5, Process Description
6.8 Product Handling	For the product or products resulting from operation of the reprocessing plant, provide the details of the system and the methods that will be used for their handling and storage.	Section 2.5, Process Description
6.8.2.4 Safety Criteria and Assurance	Describe all safety-related features, systems, and special techniques to be employed to ensure the safety of the operation. Include systems needed for criticality control and radiological safety.	Section 2.5, Process Description
6.9 Items Requiring Further Development	Present in the PSAR the details for each item requiring the development of additional information or data. Identify the item, what is needed, and how and when the information will be obtained. Indicate times when data are expected to be available and options that are available in the event satisfactory results are not obtained.	These requirements are similar to those that would be required in DOE-STD-3009-94 ^b for Section 3.3.2.3.1 for a new facility.

Table 4. Items of Regulatory Guide 3.26^a that are Unique to Fuel Conditioning. (6 sheets)

Section and title from Regulatory Guide 3.26 ^a	Item unique to fuel conditioning	Cold Vacuum Drying Facility Safety Analysis Report section and title
8.1.2 Design Considerations	Discuss the arrangement and plans for decontamination of the plant as referred to in 10 CFR Part 50, ^c Appendix F, "Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities."	These requirements are similar to those of DOE-STD-3009-94 ^b for Chapter 16, "Provisions for Decontamination and Decommissioning."
11.5.2.2 Technical Conditions and Characteristics	Technical conditions and characteristics should be stated in terms of allowable quantities (e.g., temperature, pressure, mass of fissionable material in certain systems, concentration of radioactive material in certain systems, volume of fluid required in a system, chemical constitution of certain fluids, or allowable configurations of equipment).	These requirements are similar to those of DOE-STD-3009-94 ^b for Chapter 5, "Derivation of Technical Safety Requirements," to the extent that the requirements relate to the protection of public health and safety.

^a NRC, 1975, Standard Format and Content of Safety Analysis Reports for Fuel Reprocessing Plants, Regulatory Guide 3.26, U.S. Nuclear Regulatory Commission, Washington, D.C.

^b DOE-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.

^c 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," Code of Federal Regulations, as amended.

PSAR = preliminary safety analysis report.

Table 5. Information to be Added to the Final Safety Analysis Report for the Cold Vacuum Drying Facility that is not Required by DOE-STD-3009-94^a and that is Unique to the Conditioning Process. (3 sheets)

Section and title from DOE-STD-3009-94 ^a	Material to be included
Executive Summary and Section 2.3, Facility Overview	Describe the type and exposure of the fuel to be conditioned.
Section 2.5, Process Description	<p>Provide a detailed description of the physical, chemical, and radiological characteristics of the spent fuel to be conditioned or provide a reference for this information. Include feed specifications, such as fissile material limits, fission product concentrations, forms of the material, packaging, and irradiation history.</p> <p>Include a summary description of the conditioning process to be used, including process origin and background. Identify process capacity, basic material flow and balance for the feed and product, and waste streams. For the product or products resulting from operation of the conditioning facility, provide the details of the system and the methods that will be used for their handling and storage.</p> <p>Discuss support systems, the interfaces between these systems, ancillary activities, and the main process.</p> <p>Identify the products, including the product characteristics and specifications, and byproducts that result from the plant operation. Provide information related to onsite waste processing, waste disposal or holding areas, transportation, and utility and water supplies.</p> <p>Describe the general plant conditions and surveillance needs in various shutdown modes--extended, short-term, and emergency. Indicate the time required to shut down and start up for each mode. Discuss how a control room and/or control areas are to be designed to permit occupancy and actions to be taken to operate the plant safely under normal conditions and to maintain the plant in a safe condition under accident or other abnormal conditions.</p> <p>Discuss the process design bases, including materials of construction, pressure and temperature limits, and detailed dimensions.</p> <p>Provide a summary description of the instruments used to detect process conditions and the systems used to control the process. For safety class instruments, the description should include testability, redundancy, and failure conditions.</p> <p>Discuss the instrumentation and control features associated with process control, process monitors, and alarms, and the relationship of one to the other. Identify aspects that must be relied upon to ensure continued safe operation or safe shutdown under accident conditions. Describe the redundancy of any safety-class features. Discuss the provisions for in situ testability of any safety-class instrumentation and control systems. Describe how such instrumentation and control systems are designed to be fail-safe or to assume a state demonstrated to be acceptable if conditions such as disconnection, loss of energy or motive power, or adverse environments are experienced.</p> <p>Describe in detail installation of spare or alternative safety-class instrumentation designed to provide continuity of operation under normal and abnormal conditions. Also, describe the bases for inspection, preventive maintenance, and testing programs to ensure continued safe functioning.</p> <p>Discuss provisions for obtaining samples for process analysis and control necessary to ensure that operations are within prescribed limits. Describe the facilities and analytical equipment that will be available to perform the analyses, as well as the destination of laboratory wastes.</p> <p>Provide assurance that sharing of common utilities and services and physical interaction between the CVDF and other facilities do not impair the capability of either facility to perform its safety functions (Note: The consideration for sharing of structures is to be discussed in Section 2.4, "Facility Structure.") (Garvin 1996)^b</p>

Table 5. Information to be Added to the Final Safety Analysis Report for the Cold Vacuum Drying Facility that is not Required by DOE-STD-3009-94^a and that is Unique to the Conditioning Process. (3 sheets)

Section and title from DOE-STD-3009-94 ^a	Material to be included
Section 2.6, Confinement Systems	<p>Discuss each method of confinement that will be used to ensure that there will be no uncontrolled release of radioactivity to the environment. Include for each</p> <ol style="list-style-type: none"> 1. Criteria for protection against any postulated internal accident or external natural phenomena 2. Design criteria selected for vessels, piping, effluent systems, and backup confinement 3. Delineation of the extent to which the design is based on achieving the lowest practical level of releases from the operation of the plant. <p>Where the limits selected are consistent with proven practice, a referenced statement to that effect will suffice; where the limits extend beyond present practice, an evaluation and an explanation based upon developmental work and/or analysis should be provided. Those criteria may be expressed as explicit numbers or as general conditions.</p>
Section 4.3, Safety-Class Systems, Structures, and Components	<p>Demonstrate implementation of the NRC nuclear safety equivalency requirements documented in WNC-SD-SNF-DB-003.^b</p> <p>If the CVDF includes safety-class electrical and/or instrumentation and control systems, commit to using IEEE Standard 603-1991.^c</p> <p>If the safety-class electrical or instrumentation and control systems also use safety-class lead storage batteries, commit to using IEEE Standard 484-1987^d and IEEE Standard 535-1986.^e</p> <p>If the CVDF includes safety-class electrical equipment that may be exposed to harsh environments during off-normal or postulated accident conditions, commit to the additional environmental qualification requirements identified in WNC-SD-SNF-DB-003.</p> <p>If the CVDF uses safety-class instrument air, motor-operated valves, or an open-cycle cooling system, document the use of NRC Generic Letters 88-14,^f 89-10 (with supplements),^g and 89-13^h as appropriate.</p>
Section 4.4, Safety-Significant Structures, Systems, and Components	<p>State that the requirements of ASME N509-1989ⁱ and ASME N510-1989^j have been incorporated into the design of safety-significant heating, ventilating, and air conditioning systems used to achieve onsite radiological limits or to implement ALARA principles (Garvin 1996).</p>
Chapter 6.0, Prevention of Inadvertent Criticality	<p>Include a commitment to a criticality safety k_{eff} value of 0.95 (Garvin 1996).^b</p>
Chapter 7.0, Radiation Protection	<p>State that the CVDF has incorporated control devices for access to high-radiation areas that conform to the requirements of 10 CFR 20.1601, "Control of Access to High Radiation Areas."^k 10 CFR 20.1601^l requires control devices for all high-radiation areas that meet the definition in 10 CFR 20.1003, "Definitions,"^k of 0.1 rem in 1 hour at 30 cm, whereas 10 CFR 835^m does not require incorporation of control devices until the dose rate in an accessible area reaches 1 rem in 1 hour at 30 cm. Control devices are hardware features, such as alarms or locked entryways, as opposed to administrative controls. Also state that the CVDF has incorporated into the design for any unrestricted areas the 10 CFR 20.1301, "Dose Limits for Individual Members of the Public,"^k hourly dose limit of 0.002 rem to the public from external sources during normal operations and anticipated occurrences (Garvin 1996).^b</p>
Section 7.4, ALARA Policy and Program	<p>Show that the design requirements of Regulatory Guide 8.8ⁿ have been implemented.</p>
Section 7.7, Radiological Monitoring	<p>Document that the provisions for radiological monitoring will meet the requirements of 10 CFR 20^k and 10 CFR 835^l (Garvin 1996).^b</p>

Table 5. Information to be Added to the Final Safety Analysis Report for the Cold Vacuum Drying Facility that is not Required by DOE-STD-3009-94^a and that is Unique to the Conditioning Process. (3 sheets)

Section and title from DOE-STD-3009-94 ^a	Material to be included
Chapter 8, Radiation Protection	Discuss the provisions provided in the facility to limit radiological exposure by shielding.
Section 11.4, Fire Protection	According to WMC-SD-SNF-DB-002, Table 5.b, Section 50.48, and Appendix R, ⁿ the final design of new SNF Project Path Forward facilities should be reevaluated to reconfirm that DOE Orders 5480.7A ⁿ and 6430.1A ⁿ provide adequate fire protection requirements to achieve nuclear safety equivalence. An aspect of the design to be reconfirmed is the lack of active safety-class prevention or mitigation systems. Address the review and the additional requirements in the FSAR if it is determined that the requirements of DOE Orders 5480.7A ⁿ and 6430.1A ⁿ need to be augmented by the NRC requirements discussed above.
Chapter 13.0, Human Factors	According to WMC-SD-SNF-DB-003, ^b the CVDF project was to review the NRC guidance of NUREG-0700 ^d and Standard Review Plan 18.1 ¹ against DOE Order 6430.1A, ⁿ Section 1300-12.4, to identify appropriate additional NRC guidance for design of the facility. The review was to consider the differences in complexity between a power reactor control room and that of the CVDF. In the FSAR, discuss the results of this review and the extent to which the requirements of NUREG-0700 ^d and Standard Review Plan 18.1 ¹ have been incorporated into the design of the CVDF (Garvin 1996). ^c

^aDOE-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.

^bGarvin, L. J., 1996, Spent Nuclear Fuel Project Path Forward, Additional NRC Requirements, WMC-SD-SNF-DB-003, Rev. 2, Westinghouse Hanford Company, Richland, Washington.

^cIEEE, 1991, Standard Criteria for Safety for Nuclear Power Generating Stations, IEEE Standard 603-1991, Institute of Electrical and Electronic Engineering, Piscataway, New Jersey.

^dIEEE, 1987, Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations, IEEE Standard 484-1987, Institute of Electrical and Electronic Engineering, Piscataway, New Jersey.

^eIEEE, 1986, Qualifications of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations, IEEE Standard 535-1986, Institute of Electrical and Electronic Engineering, Piscataway, New Jersey.

^fNRC, 1988, Instrument Air Supply System Problems Affecting Safety-Related Equipment, Generic Letter 88-14, U.S. Nuclear Regulatory Commission, Washington, D.C.

^gNRC, 1989, Safety-Related Motor-Operated Valve Testing and Surveillance, Generic Letter 89-10, U.S. Nuclear Regulatory Commission, Washington, D.C.

^hNRC, 1989, Service Water System Problems Affecting Safety-Related Equipment, Generic Letter 89-13, U.S. Nuclear Regulatory Commission, Washington, D.C.

ⁱASME, 1989, Nuclear Power Plant Air-Cleaning Units and Components, ASME NS09, American Society of Mechanical Engineers, New York, New York.

^jASME, 1989, Testing of Nuclear Air Treatment Systems, ASME NS10 (R95), American Society of Mechanical Engineers, New York, New York.

^k10 CFR 20, "Standards for Protection Against Radiation," Code of Federal Regulations, as amended.

^l10 CFR 835, "Occupational Radiation Protection," Code of Federal Regulations, as amended.

^mNRC, 1982, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonably Achievable, Regulatory Guide 8.8, Rev. 1 (proposed), U.S. Nuclear Regulatory Commission, Washington, D.C.

ⁿGarvin, L. J., 1996, Spent Nuclear Fuel Project Path Forward, Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities, WMC-SD-SNF-DB-002, Rev. 1, Westinghouse Hanford Company, Richland, Washington.

^oDOE Order 5480.7A, 1993, Fire Protection, DOE 5480.7A, U.S. Department of Energy, Washington, D.C.

^pDOE Order 6430.1A, 1989, General Design Criteria, DOE 6430.1A, U.S. Department of Energy, Washington, D.C.

^qNRC, 1995, Human System Interface Design Review Guidelines, NUREG-0700, Rev. 1 (for comment), U.S. Nuclear Regulatory Commission, Washington, D.C.

^rNRC, 1981, Standard Review Plan, Section 18.1, "Human Factors Engineering, Control Room," NUREG-0800, U.S. Nuclear Regulatory Commission, Washington, D.C.

ALARA = as low as reasonably achievable.

CVDF = Cold Vacuum Drying Facility.

FSAR = final safety analysis report.

NRC = U.S. Nuclear Regulatory Commission.

SNF = spent nuclear fuel.

This page intentionally left blank.

DISTRIBUTION SHEET

To	From	Page 1 of 1
Distribution	L. J. Garvin/Nuclear Safety	Date 05/07/97
Project Title/Work Order		EDT No. 607687
Additional Guidance for Including Nuclear Safety Equivalency in the Canister Storage Building and Cold Vacuum Drying Facility Final Safety Analysis Report.		ECN No. N/A

Name	MSIN	Text With All Attach.	Text Only	Attach./Appendix Only	EDT/ECN Only
G. D. Benecke	R3-15	X			
D. Bergman	X3-79	X			
D. M. Black	S0-04	X			
R. D. Crowe	R3-15	X			
P. L. Davis	X3-79	X			
R. P. DiPiazza	R3-15	X			
L. Fernandez	R3-15	X			
B. R. Fillion	B7-41	X			
L. J. Garvin	R3-15	X			
T. D. Hays	B7-41	X			
M. J. Higuera	S0-04	X			
A. G. Hosler	H0-50	X			
M. A. Jensen	X3-79	X			
R. J. Kidder	A2-25	X			
D. E. Krahn	R3-15	X			
M. Kummerer	R3-15	X			
P. G. LeRoy	R3-15	X			
B. D. Lorenz	R3-15	X			
R. Meichle	X3-79	X			
T. D. Merklng	X3-79	X			
R. P. Omberg	R3-15	X			
D. A. Reny	R3-15	X			
D. L. Scott	A3-34	X			
D. A. Smith	R3-15	X			
SNF Project Files	R3-11	X			
Central Files	A3-88 (2)	X			
CSB file	SB-06	X			